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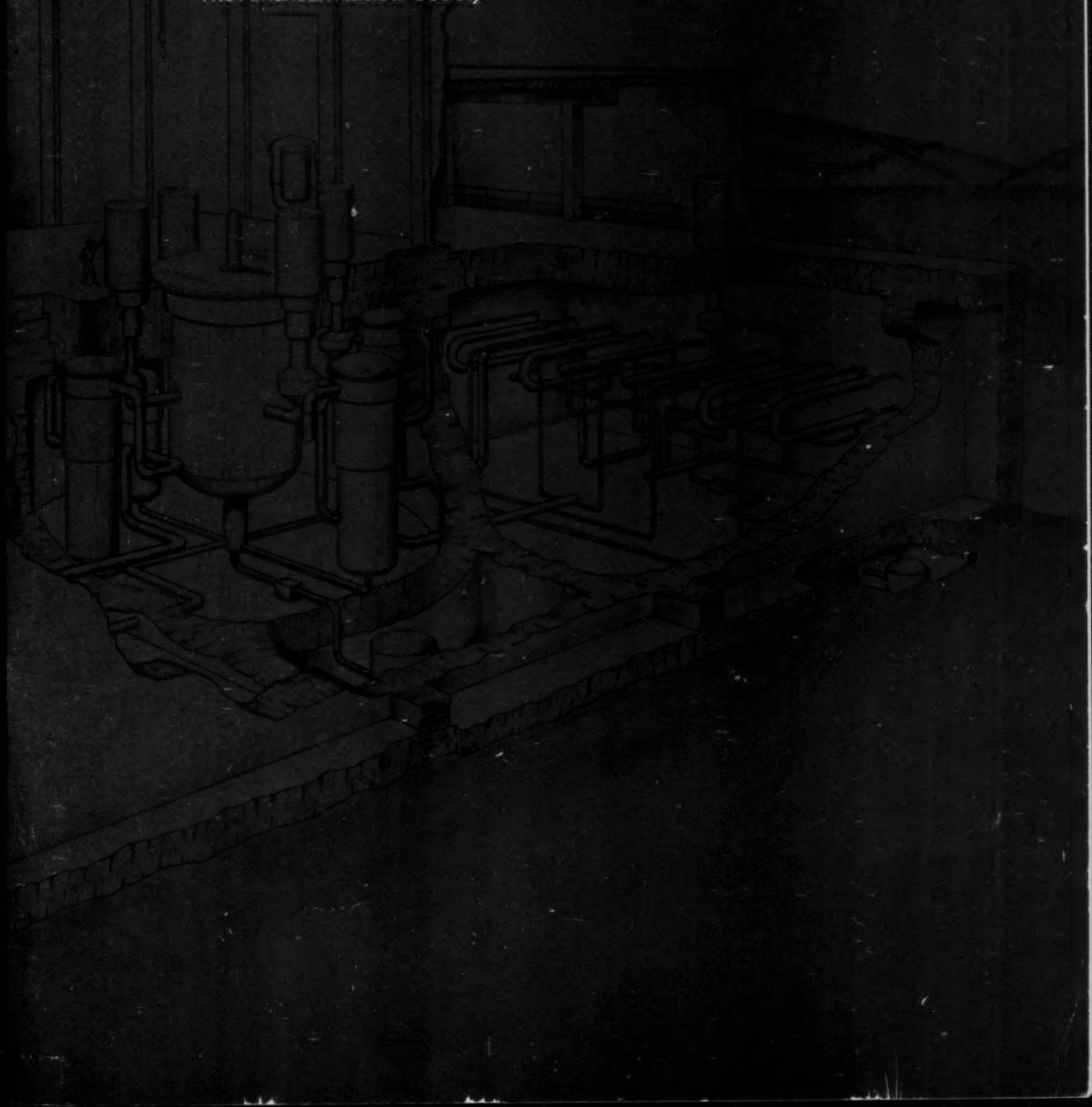

TECHNICAL PROGRESS REVIEWS

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REVIEW ARTICLES

AMERICAN NUCLEAR SOCIETY—CRITICAL REVIEWS

See page 366 for the index to Volume 14.

CURRENT AWARENESS REVIEWS

Myrna L. Steele, who has been preparing the Current Awareness Reviews, has accepted a new assignment with the Division of Operational Safety at AEC Headquarters. Because of this move, Miss Steele was unable to complete the reviews intended for this issue. She intends to contribute to the journal, as time permits, in the future on a reduced scale.

The editors wish to express their appreciation for Miss Steele's past contributions to the journal and wish her success in her new undertaking.

COVER: A conceptual drawing (from Oak Ridge National Laboratory) of a molten-salt breeder reactor. See the article on page 335.

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REACTOR TECHNOLOGY

is a quarterly review of progress and developments in the reactor field including the following subject areas:

Economics	Fuel Cycles
Physics	Fluid and Thermal Technology
Mechanics	Fuel Processing
Construction	Components
Fuel Elements	Operating Performance

Prepared for reactor physicists, reactor operators, designers, fuel-cycle specialists, administrators, and others interested in the field, this journal reviews, summarizes, critically evaluates and interprets the state of the art as reflected in the current literature.

The many reports and publications referenced merit further study, and readers are urged to consult them for additional details.

Qualified authors are invited to contribute review articles for consideration for publication.

U. S. Atomic Energy Commission
Division of Technical Information
Washington, D. C. 20545

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The Technology of In-Pile Creep Measurements

By Peter Cybulskis*

Abstract: *The possible effects of radiation-induced creep on the long-term performance of fuel and structural materials have been a source of continuing concern in the development of power reactors. Owing to the extremely severe thermal and radiation environments to be encountered, radiation-induced creep is of particular concern in the design of liquid-metal-cooled fast breeder reactors (LMFBRs). However, this concern is by no means limited to LMFBRs. A number of experimental programs have been undertaken to investigate the nature of the enhanced plasticity observed to take place in reactor environments. These programs encompass ceramic fuel as well as metallic cladding and structural materials under a variety of conditions. The experimental phases of the various programs are reviewed, with particular emphasis on the technology of specimen and capsule design, method of specimen loading, environmental control, and nature and precision of experimental measurements. Representative results are shown, as appropriate, to indicate the degree of success that each approach has achieved.*

Irradiation in a nuclear reactor increases the creep rates of ceramic fuels as well as of metallic cladding and structural materials. Such increased creep can lead to fuel-diameter increases, warpage of fuel pins, and changes in the dimensions and geometry of coolant passages. These effects in turn can precipitate undesirable temperature and reactivity changes in the core which may affect the lifetime and safety of the system. Because of the extremely severe thermal and radiation environments to be encountered, radiation-induced creep is of particular concern in the development of liquid-metal-cooled fast breeder reactors (LMFBRs). As a result of this concern, a number of programs have been or are being undertaken to investigate the nature and extent of the enhanced plasticity that occurs in a radiation environment. In a previous issue of *Reactor Technology*,[1] Gilbert has reviewed the theories and analytical models that were developed to describe the effect of radiation on the creep of materials. This article reviews the various experimental programs on radiation-induced creep, with particular emphasis on the technology of specimen and capsule design, method of specimen loading, environmental control, and nature and precision of experimental measurements. Representative results are given, as appropriate, to indicate the extent of success attained by a particular experimental approach.

The basic requirements for creep testing are the application of a constant stress to a test specimen that is maintained at a constant temperature and the measurement of the specimen strain as a function of time. As would be expected, a number of approaches for meeting these requirements have been suggested and tried.

CREEP OF CLADDING AND STRUCTURAL MATERIALS

Uninstrumented Experiments

Helical-Specimen Tests at Dounreay. Uninstrumented in-pile creep experiments on metallic cladding and structural materials were performed in the Dounreay Materials Testing Reactor.[2] The specimens were made from molybdenum, nickel, titanium, Nimonic alloys PE16 and 80A, and 316 stainless steel, in the form of 0.5-in.-OD helical springs wound from 0.04-in.-diameter wire having 25 active turns. For these specimen dimensions a spring deflection of 0.05 in. corresponds to a surface shear strain of 10^{-4}. The tops

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and bottoms of the springs were screwed into, and spot welded to, threaded connections that provided measurement surfaces. Twelve specimens in two strings of six each were hung under a tensile load in a D_2O-flooded thimble. The temperature of the springs, which was estimated at about 100°C, was determined by gamma heating on the one hand and boiling of the surrounding D_2O on the other. After each reactor shutdown the springs were hoisted into a portable shielded cell and photographed through lead-glass windows. From the photographs, showing reference surfaces above and below each spring and an adjacent scale, spring extensions could be determined to ±0.3 mm.

Helical springs loaded in tension have also been used to investigate in-pile creep in the Dounreay Fast Reactor (DFR).[3] The specimens, essentially identical to those described above for thermal irradiations, were suspended in strings of five inside a sodium-filled and sealed stainless-steel capsule. The springs were fitted at top and bottom with reference measuring surfaces. Weights were added at various positions in the specimen train to cover the stress range of interest; the weights also contained fusible-alloy temperature monitors. Creep deflections were determined from preirradiation and postirradiation X radiographs to an estimated accuracy of ±0.25 mm. The indicated temperature range was 230 to 300°C. A number of these assemblies have been successfully irradiated, and additional exposures are under way.

Pressurized-Tube Tests at Argonne National Laboratory. Perhaps the most straightforward in-pile creep experiments encountered are those performed by Argonne National Laboratory (ANL)[4] with sealed, prepressurized, unfueled cladding tubes irradiated in EBR-II. The experiment consisted in irradiating a Mark B37 subassembly containing 18 pressurized annealed 304L stainless-steel tubes exposed directly to the flowing primary sodium. These tubes had a 0.290-in. outside diameter, a 0.020-in. wall, and a 60-in. length. They were charged with helium in groups of three to give hoop stresses of 10,000, 15,000, 20,000, 25,000, 30,000, and 35,000 psi at the 700°F sodium inlet temperature. Filler rods (0.238 in. in outside diameter) of 304 stainless steel were used inside each tube to reduce the potential energy release of an unexpected failure. The remaining 19 positions in the subassembly contained dummy tubes, rods, temperature monitors, and flux monitors. The diameters of the capsules at a number of axial locations were measured after each reactor cycle with an optical gauge; profilometer measurements over the entire capsule length were subsequently added.

After six cycles[5] the capsules had received a total fluence (>0.1 MeV) of 1.2×10^{22} fast neutrons/cm^2. The diameter-growth data were analyzed to determine the dependence of the neutron-flux-enhanced creep rate on hoop stress. For the pressurized capsules, both swelling and creep contributed to the observed hoop strain. The hoop strain of the capsules that were not pressurized represents the diameter change due to swelling, and this was subtracted from the total strain determined for the pressurized capsules. The remaining hoop strain is, then, the strain due to neutron-flux-enhanced creep. The enhanced creep strain was plotted as a function of time (or fluence) at each stress level, and the creep rate was calculated from the slope of the plotted line. Figure 1 shows the enhanced hoop strain for each value of hoop stress as a function of time at power, or fluence, for the initial experiment. Flux is given as 10^{15} neutrons/(cm^2)(sec) (>0.1 MeV). Each point in this figure represents the average value of the creep hoop strain for the capsules at that particular stress. Figure 2 shows the neutron-flux-enhanced creep rate ($\dot{\epsilon}_H$), hr^{-1}, as a function of hoop stress (σ_H), ksi, based on the assumption of a linear time dependence of the creep strain. The following expression represents the linear least-squares fit to the data shown in Fig. 2:

$$\dot{\epsilon}_H = 1.5 \times 10^{-11}\, \sigma_H + 2.8 \times 10^{-8} \quad \text{hr}^{-1}$$

A major uncertainty in the interpretation of these results is believed to be related to the temperatures and temperature gradients existing in the subassembly.[6] A progressive increase in diameter from the bottom to the top of the capsules is said to be the result of a temperature gradient due to radial heat conduction from adjacent subassemblies. The initial attempt at measuring temperatures with melt wires was unsuccessful. The results of subsequent additions of silicon carbide and sodium-expansion temperature monitors are not yet available.

Instrumented Experiments

Tensile-Creep Tests at Chalk River. Uniaxial in-reactor creep measurements[7,8] on various zirconium alloys have been conducted by Atomic Energy of Canada Limited at Chalk River since 1960. The original specifications called for a creep machine suitable for insertion into a reactor fast-neutron rod with a 1.187-in. bore, capable of holding and stressing the tensile specimen remotely and maintaining a constant

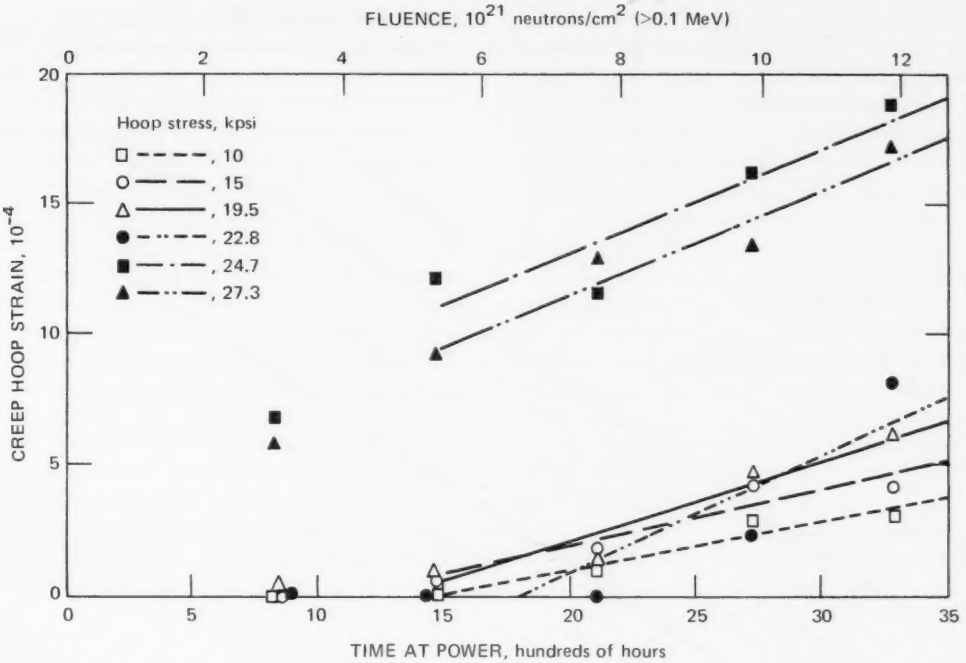

Fig. 1 Radiation-enhanced creep strain as a function of fluence and time at power for helium-pressurized 304L stainless-steel capsules; irradiation temperature, 380°C; flux (>0.1 MeV), ~10^{15} neutrons/(cm^2)(sec); ANL results.[5]

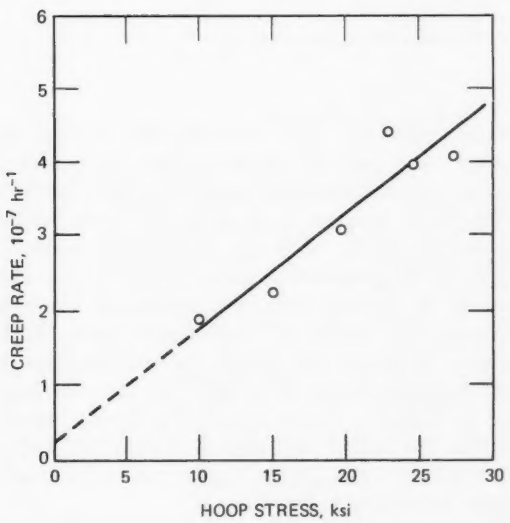

Fig. 2 Radiation-enhanced creep rate as a function of hoop stress of helium-pressurized 304L stainless-steel capsules; irradiation temperature, 380°C; flux (>0.1 MeV), ~10^{15} neutrons/(cm^2)(sec); ANL results.[5]

specimen temperature of 300 ± 4°C. Subsequently the specifications were revised to require a constant temperature within the range of 200 to 450°C in the NRX reactor [gamma heating rate of ~0.3 watt/g, flux of 1×10^{13} fast neutrons/(cm^2)(sec)] and 280 to 400°C in the NRU reactor [gamma heating of about 3 watts/g, flux of 5×10^{13} fast neutrons /(cm^2)(sec)]. Also, the machine had to withstand a fluence on the order of 5×10^{20} fast neutrons/cm^2 and be fitted with a remotely operated strain-measuring device with a sensitivity of 50 μin. to measure axial creep at a rate of 10^{-4} to 10^{-5}%/hr over a nominal range of 0.030 in. The stress had to be held constant within ±2% in the range 20,000 to 30,000 psi.

The in-pile creep capsule that was developed was 1.125 in. in diameter, 25 ft long, and consisted of two separable assemblies; the outer assembly was the heater for maintaining the desired temperature, and the inner assembly was the creep experiment itself. The Mark IV H design as used in the NRX reactor is shown schematically in Fig. 3. The specimen was stressed by an externally pressurized bellows that was calibrated so that it served to both apply and measure the load. The

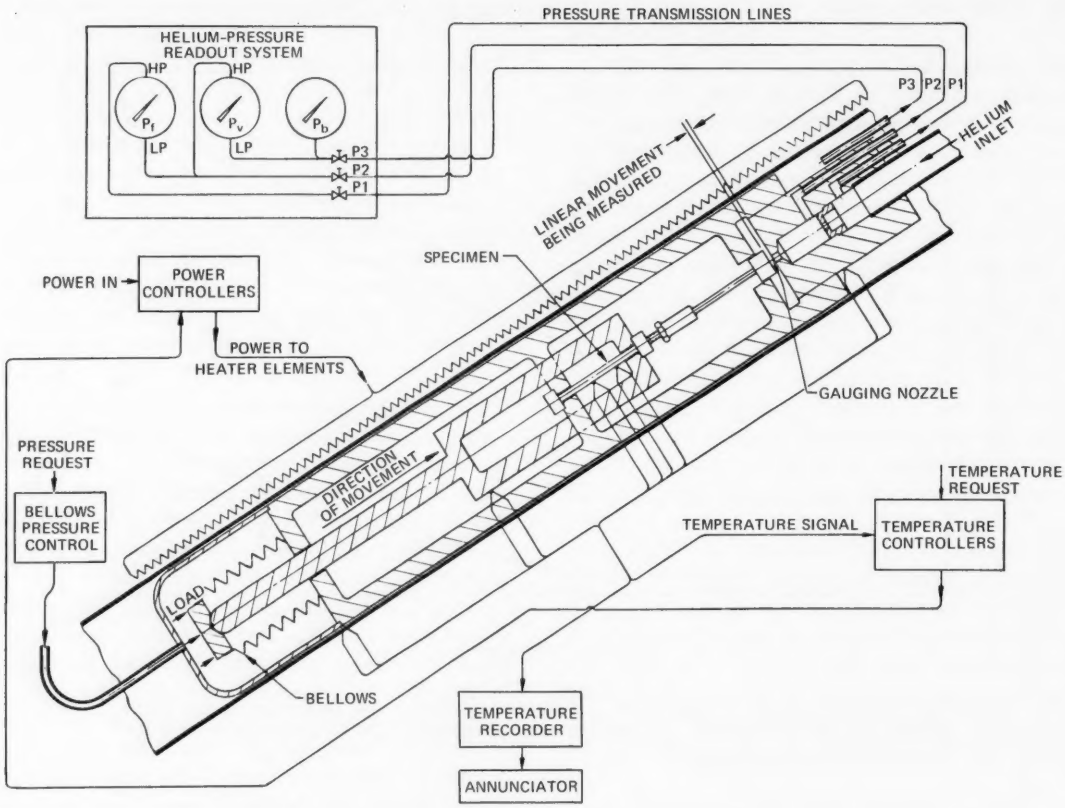

Fig. 3 Schematic diagram of NRX Mark IV H in-reactor creep machine.[8]

bellows was contained in a pressure vessel and mechanically arranged so the bellows element shortened with increasing pressure (load). The load was transmitted to the specimen by two rectangular yokes that lay in planes perpendicular to each other. The specimen had tee-shaped ends, with an increased diameter between the test section and the ends, and was located in shaped slots in the fixed and moving crossheads. The moving or upper end of the specimen was connected to an extension rod that carried the baffle of a pneumatic nozzle used for measuring strain. The operating gap between the nozzle orifice and the baffle was adjusted at the start of a test by a threaded connection between the specimen and the extension rod.

The pneumatic nozzle consisted of a chamber housing two orifices; one orifice was located at the front of the chamber and the other at about midlength. The nozzle baffle was located immediately before the front orifice at a predetermined gap established by the initial calibration. Gas flow was fed through the orifices and past the baffle, and the pressure drop across each orifice was measured. For a given gas flow and fixed pressure drop across the inner orifice, movement of the baffle due to specimen extension changed the pressure drop across the front orifice. This change in pressure drop was related to specimen extension by means of a calibration curve. Baffle shapes can be changed to vary the effective range of the pneumatic nozzle. Flat baffles are used for ranges up to 0.030 in. with resolution of 10 μin.; needle baffles can extend the range to 0.3 in., but with a reduced sensitivity. The development of the pneumatic gauge is described in Ref. 9.

Difficulties with temperature control in the early creep experiments led to changes in thermocouple location and a switch from nitrogen to helium as the operating gas for the capsule. Initially, swaged copper-core, aluminum-insulated, stainless-steel-sheathed

heater leads and iron–constantan thermocouples with magnesium oxide insulation and stainless-steel sheaths were used. Subsequently the thermocouples were changed to Chromel–Alumel, and the sheath material for both heater leads and thermocouples was changed to Inconel. With the Mark IV design in the NRX reactor, the specimen could be maintained within ±3°C of the specified temperature within the range 180 to 450°C. The scatter in the measured specimen extension was, in general, ±25 μin. from the mean curve and decreased to ±10 μin. at steady reactor power.

The successful development and operation of the Mark IV creep machine in the NRX reactor led to the subsequent design of a similar device for operation in the higher neutron and gamma fluxes of the NRU reactor. To prevent the specimen from overheating and to minimize specimen temperature gradients in the NRU reactor environment required that the heat transfer from the specimen to the coolant be improved and that cooling coils for operation at full reactor power be provided. The design that was developed, Mark VI, is shown schematically in Fig. 4. The concept is similar to the previous design, but the specimen location and the method of loading have been changed. The specimen was contained within the bellows chamber that was attached to the pneumatic nozzle. The bottom end of the specimen was locked into the lower plug of the chamber, and the upper end was held by the face end of the bellows. The bellows tended to shorten under the applied external pressure and applied a tensile load to the specimen. An extension piece carrying the nozzle baffle was also attached to the free end of the bellows and extended through the inside diameter of the bellows, positioning the baffle in front of the nozzle. The Mark VI design as developed for the NRU reactor was more compact than the Mark IV version, eliminated the yoke assemblies, reduced differential expansion problems, and minimized material-mass and gamma-heating effects.

A number of problems were encountered in the development of the above-mentioned devices. Many of these were mechanical in nature and were largely

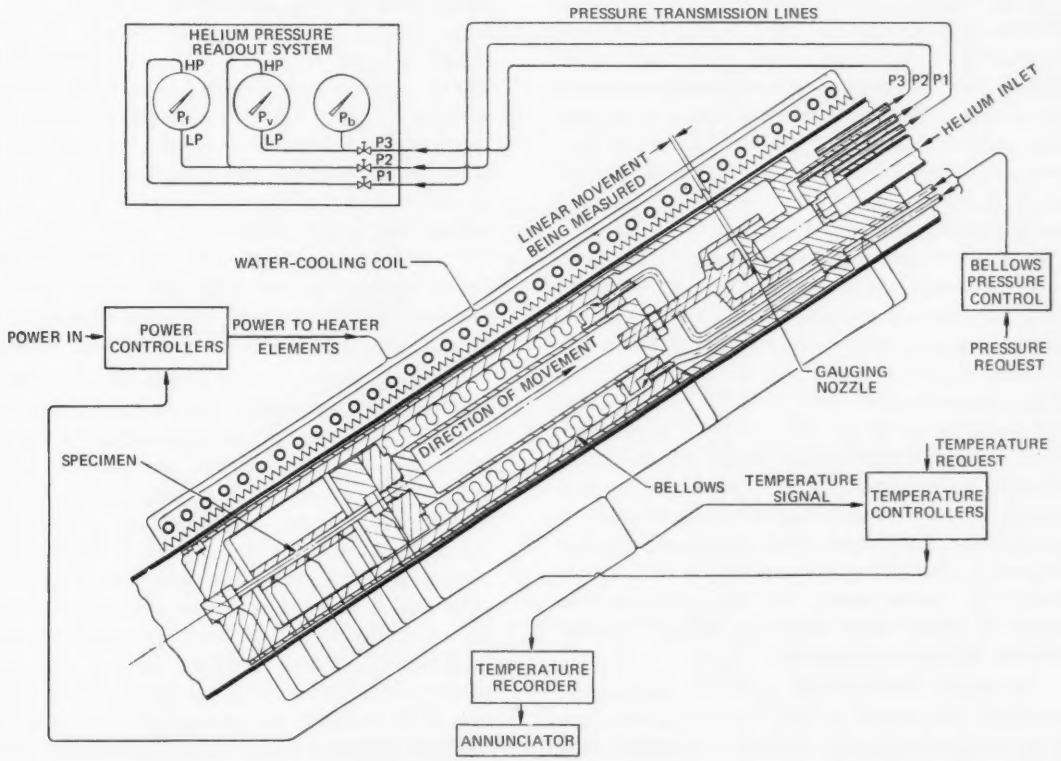

Fig. 4 Schematic diagram of NRU Mark VI in-reactor creep machine.[8]

eliminated through changes in successive experiments. Apparently the most troublesome problems had to do with overheating of various components in the creep devices and the maintenance of constant temperatures over long periods of time. Some difficulties have been encountered with flow instabilities in the pneumatic strain-measuring gauges; these difficulties were attributed to changes in the background pressure in the capsule which occurred as a consequence of changes in strain and atmospheric pressure.

The work at Chalk River has resulted in the development of in-pile creep-measuring devices that have operated successfully in fast fluxes of up to 5×10^{13} neutrons/(cm^2)(sec) and gamma heating rates of up to 3 watts per gram of stainless steel. The temperature of the test specimen could be held constant within 3°C in the range 200 to 500°C. The pneumatic strain-measuring system that was developed had an overall sensitivity of approximately 15 μin., allowing the determination of creep rate in the 10^{-7} hr^{-1} range with 10% uncertainty. Some of the results of this work are given in Refs. 10 to 12.

DFR Cladding Creep Experiment. A somewhat different approach than the one mentioned above was adopted for in-pile cladding tensile-creep studies at the DFR.[13] The experimental setup is shown schematically in Fig. 5. Flat strip specimens with a rectangular cross section, 0.05 cm by 0.42 cm by 3.8 cm long, were used. This form of test specimen was selected for ease of duplication of the conditions of heat treatment and mechanical work representative of fuel claddings. A special twin electric heater was designed for establishing uniform temperature conditions for the specimen. Each heater consisted of metal-sheathed mineral-insulated cables mounted on a stainless-steel former and sprayed over with stainless steel. One part of the heater provided a controllable path for the removal of heat generated in the specimen, and the other established an isothermal zone in which thermocouples indicated the specimen temperature without physical contact. Electrical analog studies were performed to optimize the dimensions of the heaters and evaluate temperature distributions for various combinations of reactor and heater power. An inert-gas atmosphere within the capsule acted as the heat-transfer medium between the various components.

For safety considerations, the in-pile creep experiment had to be completely sealed and no pneumatic or hydraulic connections to out-of-pile equipment were permitted in the DFR design. Loading of the specimen and measurement of the strain were therefore controlled electrically. The load was controlled by a stepping motor inside the capsule driving a screw jack that applied a load to the specimen via a tension link incorporating a strain-gauge load cell. The load cell transmitted information to a load-control unit that drove the stepping motor so as to maintain a constant load. The strain-measuring system employed a micrometer that made electrical contact with the specimen grip when turned by the stepping motor. The micrometer was turned back one step when contact was made, and the micrometer position was measured in terms of motor steps vs. time, one step representing 3.2×10^{-3} mm. The control circuit for the micrometer was arranged so alternate contact and retraction occurred automatically.

At this writing no information on the success of this design was available.

Creep Experiments in a Water-Cooled Thermal Reactor. An in-pile creep-capsule system[14-16] that was developed for operation in the relatively low-temperature-water-coolant environment of the Hanford production reactors is shown in Fig. 6. Capsules of this design have operated successfully at temperatures of 350 to 1500°F, at specimen tensile stresses of 5,000 to 80,000 psi, and for times of up to 30,000 hr. A uniaxial tensile load was applied to the creep specimen with an accuracy of ±1% through a welded 347 stainless-steel bellows. The capsule, as well as the inside of the bellows, was initially pressurized with helium; a load was applied to the specimen as the bellows was vented, resulting in a higher pressure on the outside of the bellows. The stress on the specimen was a function of the differential helium pressure, effective area of the bellows, the spring constant of the bellows, and the cross-sectional area of the specimen.

A mechanically operated screw, called the micropositioner, in conjunction with a pair of electrical contacts was used to measure elongation of the specimen. The micropositioner was attached to one end of the specimen and was coupled to a 2000 : 1 ratio gearbox in the capsule and to a drive shaft extending out of the reactor. Specimen elongation was determined by counting the number of turns of the drive shaft required to reestablish electrical contact at the other end of the specimen. The micropositioner itself had a resolution of 10 μin., but the system as a whole had a sensitivity of about 100 μin. The temperature of the specimen was maintained with a three-zone electrical-resistance heater consisting of Nichrome V wire in alumina insulators on a stainless-steel support. The junctions of the Chromel–Alumel thermocouples

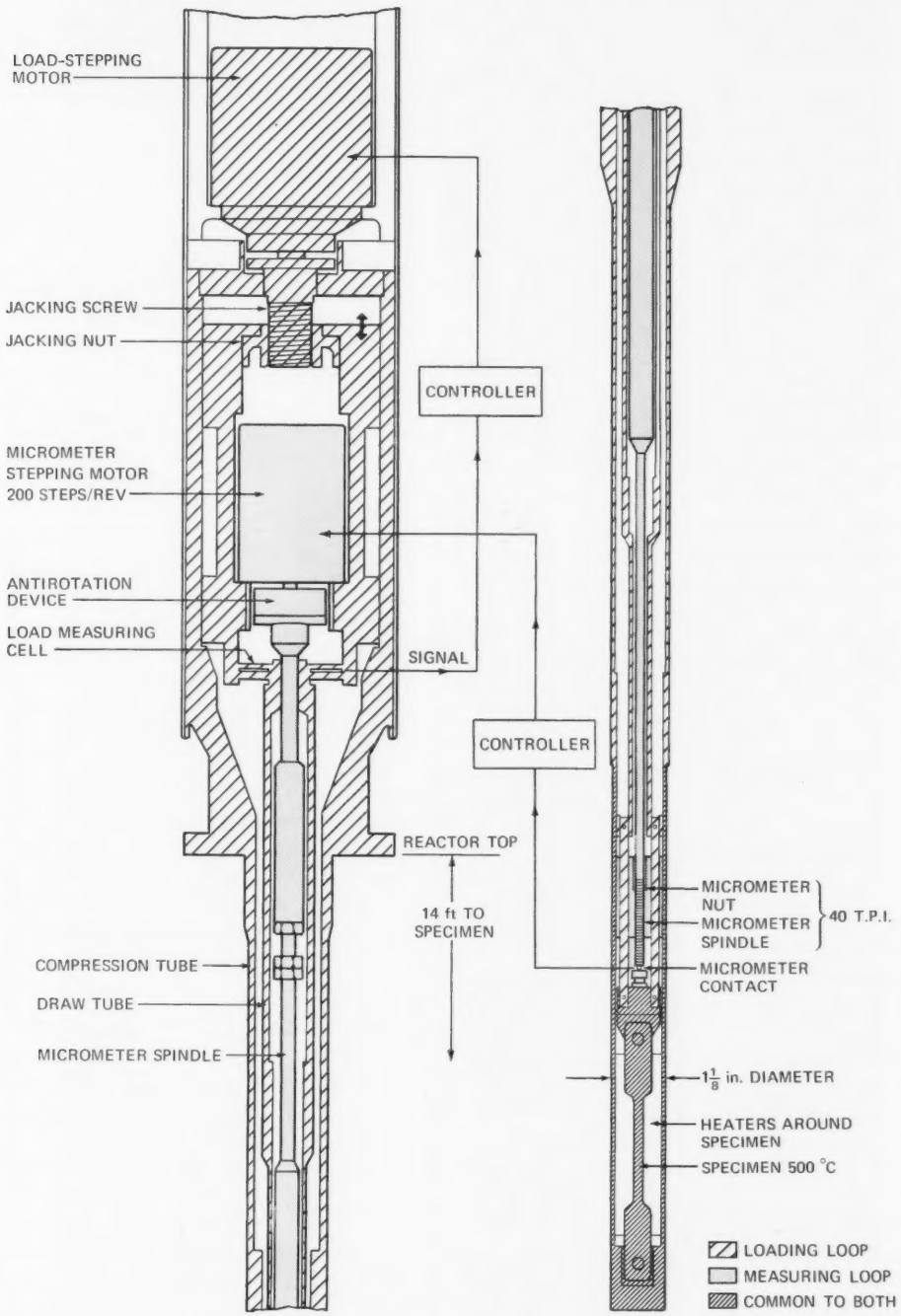

Fig. 5 The DFR cladding tensile-creep rig.[13]

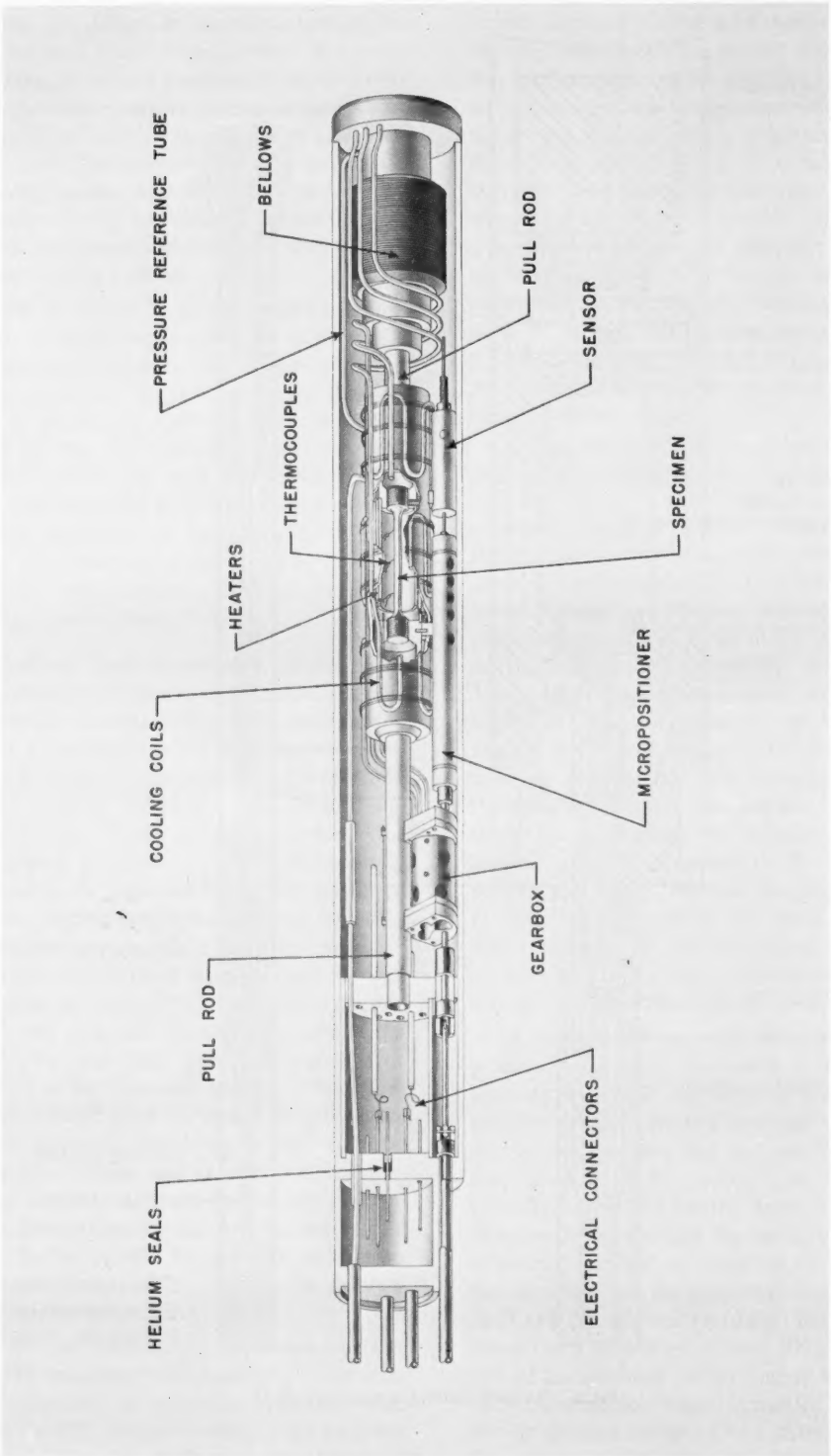

Fig. 6 Schematic of Hanford's water-cooled thermal-reactor creep capsule.[16]

used to control the heaters were held in contact with the specimen surface. During a creep test, temperatures could be held constant within ±3°F.

EBR-II In-Reactor Creep Capsule. On the basis of the successful operation of the above-mentioned thermal-reactor creep capsules, the design was extended for operation as part of an EBR-II instrumented subassembly.[16] The EBR-II creep capsule, Fig. 7, was designed to test a single uniaxial specimen with continuous in-pile strain measurement, and at the same time accommodate four prepressurized tube specimens whose strain would be measured after the irradiation. The design was aimed at operation in the 800 to 1200°F temperature range at stresses up to 20,000 psi on the stainless-steel specimens. The specimens were placed in a sodium-filled chamber that was separated from the outer capsule wall by a helium-gas gap. The temperature of the specimen chamber was maintained by a single-zone electric heater made of stainless-steel-sheathed Nichrome wire with magnesia insulation. Stainless-steel-sheathed, magnesia-insulated, Chromel–Alumel thermocouples were used to measure capsule temperatures. The system was designed to maintain temperature within ±3°F. A caged internally pressurized bellows was used to load the tensile specimen. The bellows operating temperature was expected to be 1000 to 1100°F. The specimen strain was measured by means of a mechanical system similar to the one described above. The micropositioner and gearbox were located above the specimen chamber, with the extensometer probes extending to the specimen through bellows seals. Laboratory tests were performed to verify the satisfactory performance of the strain-measuring system at the operating temperature of 1000°F. A resolution of about 100 μin. was expected for the system.

Argonne CP-5 Reactor Creep Capsule. An instrumented creep capsule was developed to test cladding materials in uniaxial tension during irradiation in the CP-5 reactor at Argonne.[17] The capsule was a 1½-in.-OD aluminum tube with a 1/16-in. wall that contained the creep machine. The creep machine was a stainless-steel tube that contained the 2-in.-gauge-length creep specimen, a fixed specimen support, a movable crosshead, pushrods, and a bellows. A linear variable differential transformer (LVDT) was fastened to the top of the inner tube; the LVDT core support was fastened to the movable crosshead. Stress was applied to the specimen by means of an internal pressure to the stainless-steel bellows that was mechanically linked to the movable crosshead by the pushrods. The pressure in the bellows could be varied from 0 to 500 psi and controlled to ±0.1 psi. Specimen stresses could range from 5000 to 100,000 psi and could be maintained within 1% of the desired value.

Specimen extension was measured by a high-temperature, radiation-resistant LVDT. A sensitivity of about 50 μin. was attained by observing a number of precautions in the use of the LVDT. Among these were: (1) locating the LVDT out of the high gamma- and neutron-flux zones, (2) making the core extension and LVDT supports out of thin-wall Invar tubing to minimize gamma heating and thermal expansion, and (3) keeping the temperatures of the LVDT as well as its power supply as constant as possible. The desired test temperature in the range 350 to 700°C was maintained by the use of sheathed Nichrome-wire heaters and/or circulating helium gas through the capsule. The outer aluminum wall of the capsule was in contact with reactor cooling water that was at 40°C. Sheathed Chromel–Alumel thermocouples were attached to the specimen outside the gauge length as well as being located near the center of the specimen. Temperature variations along the gauge length were less than 1°C from the mean temperature, which was maintained within 3°C of the desired value for extended periods of time.

CREEP OF CERAMIC FUEL MATERIALS

Creep Relaxation and Bending Studies on Uranium Oxide

Creep relaxation and bending studies have been performed in the Pluto reactor at Harwell[18] as part of a program to examine the effects of radiation on the creep properties of various ceramic fuels. Irradiations were performed at controlled temperatures up to 800°C in an assembly that could contain up to four Zircaloy-2 specimen holders with two fissile specimens in each. The specimens were in the form of rectangular bars 2.5 by 0.2 by 0.05 cm, surface ground to achieve a high degree of straightness. Rectangular slots in the Zircaloy holders accommodated the specimens, which were stressed by inserting thin metal shims between each specimen and its holder to subject it to a bending moment. The curvature of the beam was measured before irradiation and was checked after irradiation before removing the shims to verify that the specimen strain had not changed during the irradiation. The specimen relaxation was determined from the residual curvature after removal of the shims. Measurements

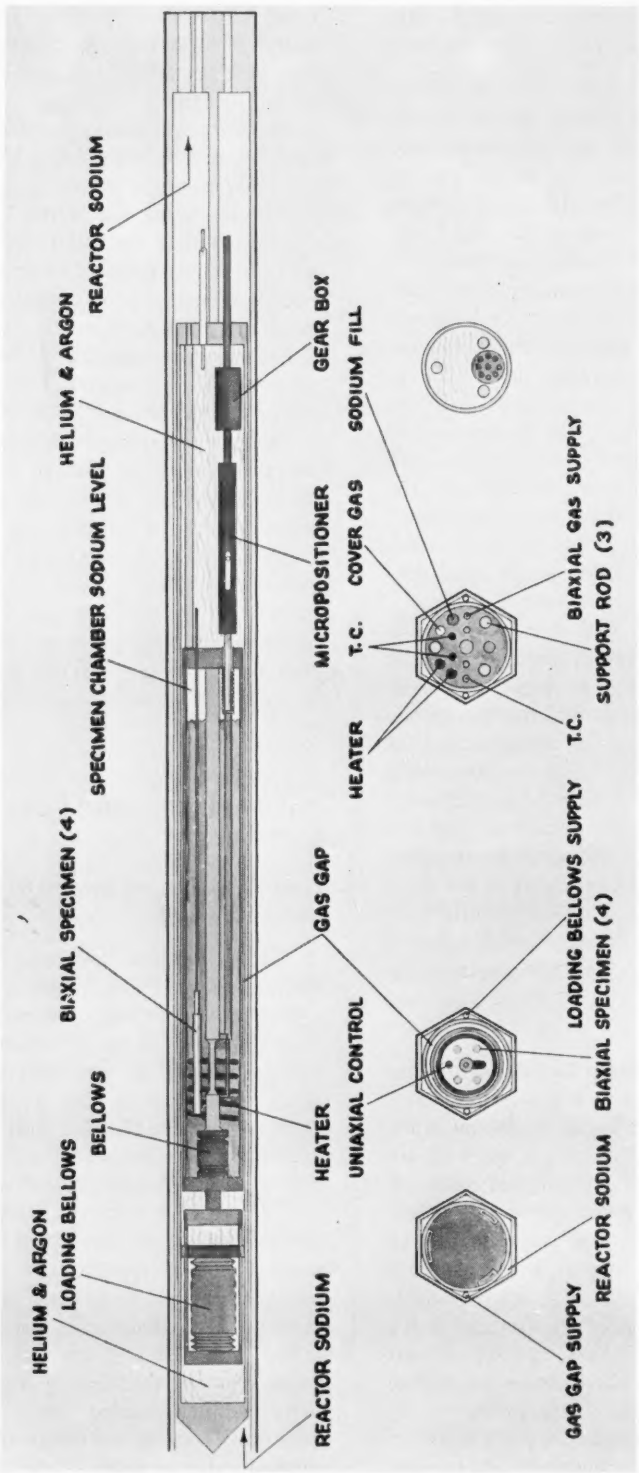

Fig. 7 EBR-II in-reactor creep capsule.[16]

were made in a shielded facility with an optical microscope and stage with an overall sensitivity of 5×10^{-4} cm. A deflection of about 5×10^{-3} cm was equivalent to a maximum fiber stress of 5000 psi. The specimen geometry and small thickness used minimized temperature gradients and thermal stresses in the specimen.

The maximum strain attainable with the above-mentioned method was limited to the elastic strain. A modified specimen holder was developed to achieve a constant-stress condition for the specimen and obtain data at higher strains. A small Nimonic 80A spring fitted into the side of the holder and acting on a Zircaloy-2 pin applied a load to the specimen that rested on alumina rollers. Creep relaxation of the spring under test conditions was negligible, but there was a reduction in specimen fiber stress with increasing specimen and spring deflection.

Irradiations have been carried out at 500 to 750°C at ratings of 0.2 to 50 watts/g on 96 to 97% dense polycrystalline UO_2 of 8- to 10-μ grain size and an initial O/U ratio of <2.0005. The results of these experiments together with those of laboratory relaxation tests are shown in Fig. 8.

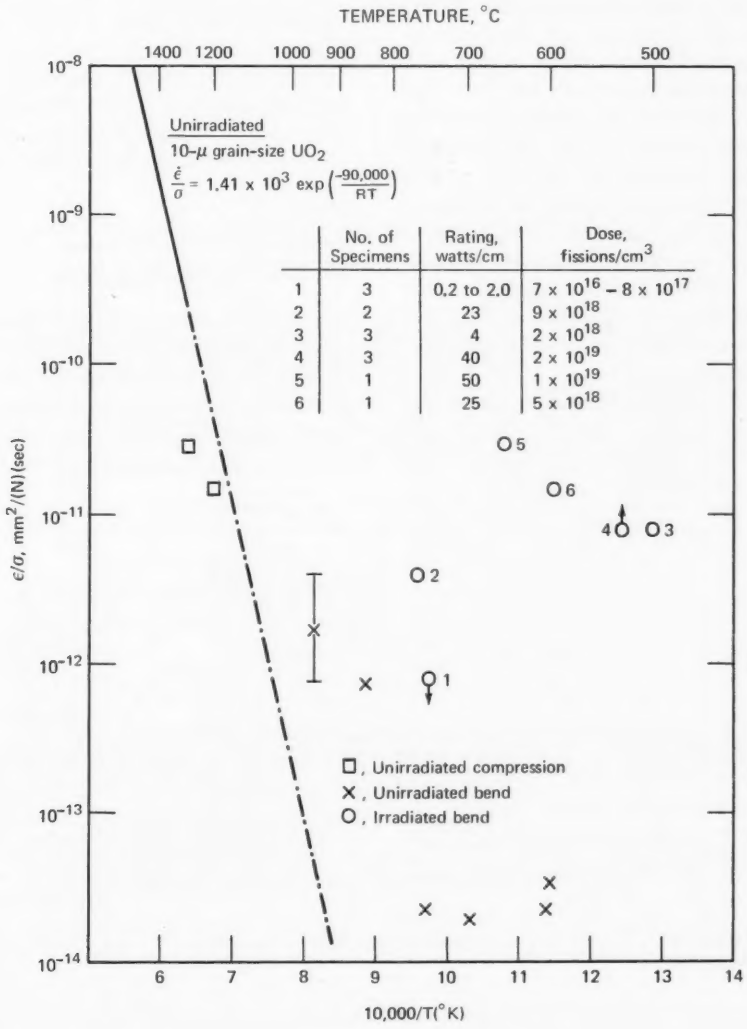

Fig. 8 Irradiation creep of UO_2—Harwell data.[18]

Tensile Creep of Uranium Carbide

The work at Harwell also includes studies on in-pile creep of UC under uniaxial tensile stress.[18,19] The first experiment involved an unenriched sintered hyperstoichiometric (4.85 wt.% carbon) uranium carbide specimen in the form of a modified Hounsfield piece, 0.317 cm in diameter with a 2.54-cm gauge length. It was held in position by split, tapered alumina collets that were located in alumina holders. One holder was attached to the capsule body and rig structure, and the other to a pull rod that extended through the in-pile section and shield plug to the top of the irradiation subassembly. A niobium spacer surrounding the specimen contained two sheathed W–Re thermocouples that were in contact with the specimen surface and two Chromel–Alumel thermocouples that measured the spacer temperature and were used for capsule temperature control. A cylindrical Inconel-sheathed heater was located outside the niobium spacer. A stainless-steel bellows forming a part of the primary containment permitted loading of the specimen. Figures 9 and 10 show the capsule configuration and overall irradiation assembly, respectively. Stress in the range of 0 to 7000 psi could be applied to the specimen by a compression spring located near the top of the rig, adjustment of the load being effected by means of a screw mechanism.

The specimen strain was measured by an in-pile cast-iron micrometer located immediately above the specimen capsule. The micrometer contact position was determined from the capacitance change between the spherical end of the micrometer shaft and an electrically insulated contact plate that formed part of the capsule end cap, using a visual-touch indicator. The micrometer reading and the applied load were displayed on adjacent scales situated at the top of the rig; these scales were read from the reactor working level using a borescope located in the reactor top biological shield plate. Both the micrometer and loading system were operated by means of removable extension shafts. In practice, it was found that the micrometer-contact position was reproducible within $\pm 2.5 \times 10^{-4}$ cm under normal conditions. Figure 11 shows the measured strain as a function of time for the UC specimen, irradiated under a stress of 3710 psi at about 800°C. The apparent strain increase of about 3×10^{-3} cm between reactor cycles is attributed to a change in the gamma-heating level in the particular reactor position with resultant changes in the measuring loop. Fracture of the specimen took place at 0.8% strain.

Compressive Creep of Uranium and Uranium–Plutonium Oxide

An in-pile creep-measuring apparatus has been developed at Battelle Memorial Institute, Columbus Laboratories,[20] for investigating the effects of the fission process on the compressive creep of ceramic fuels. The design criteria for this apparatus included requirements for: (1) a uniform specimen surface temperature of up

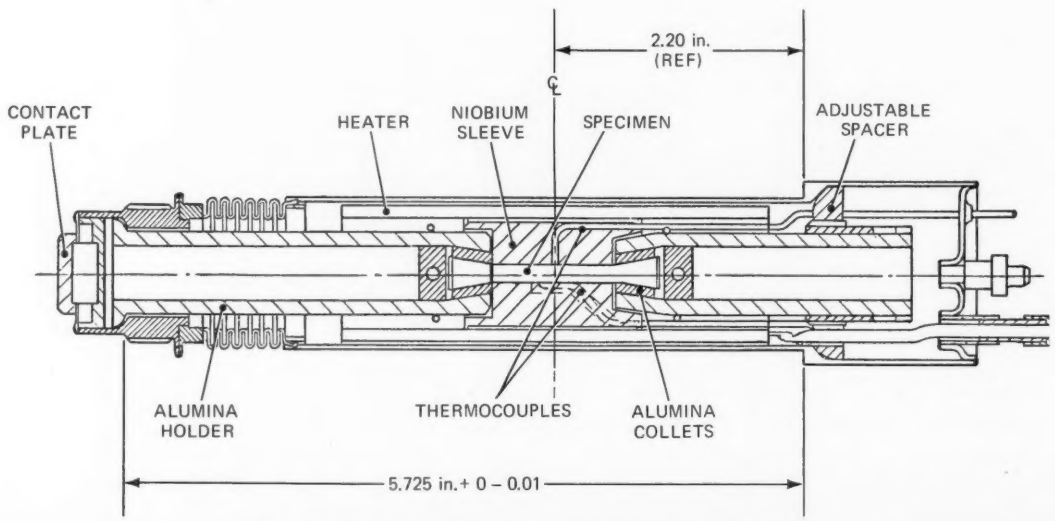

Fig. 9 Harwell capsule assembly[19] for in-pile tensile creep of UC.

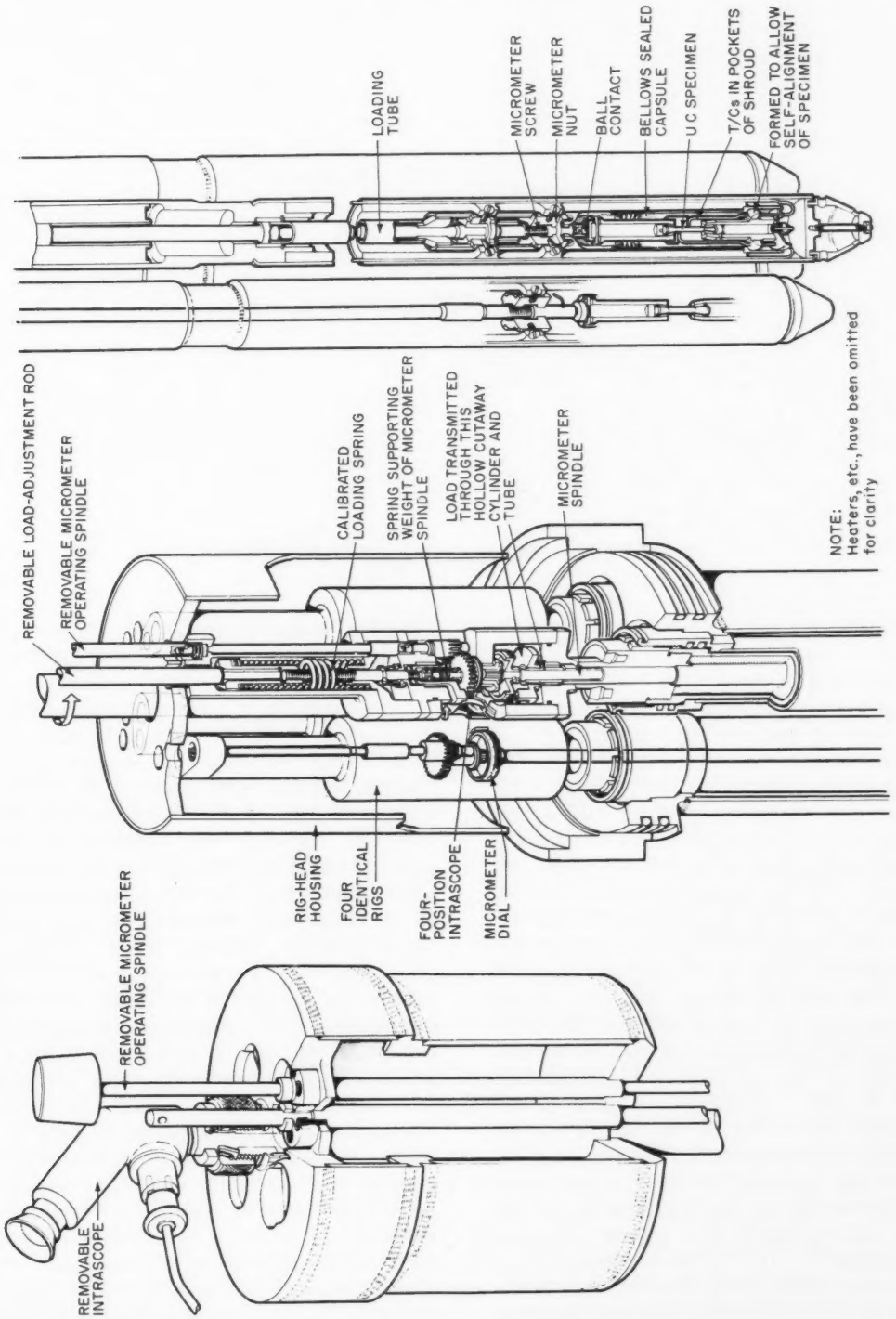

Fig. 10 Harwell irradiation assembly[18] for in-pile tensile creep of UC.

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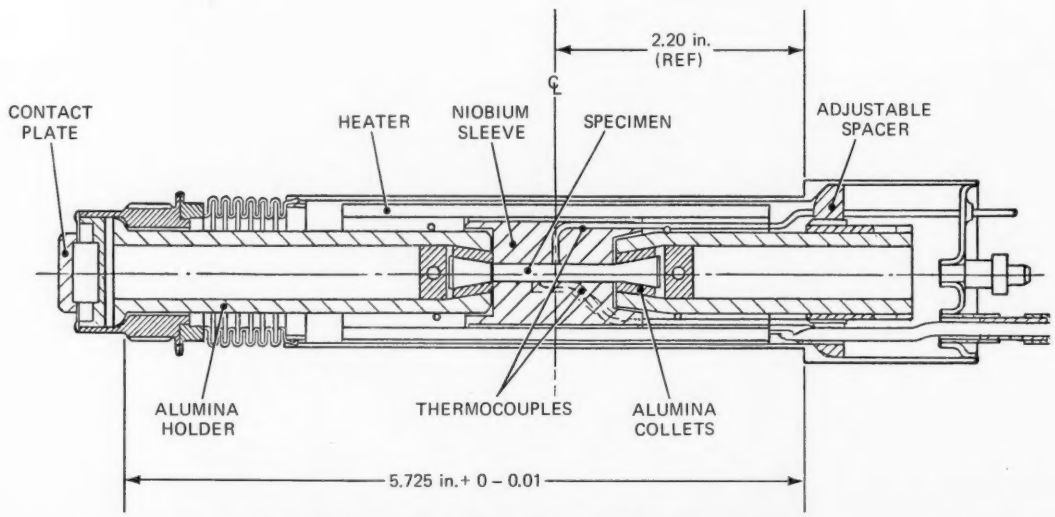

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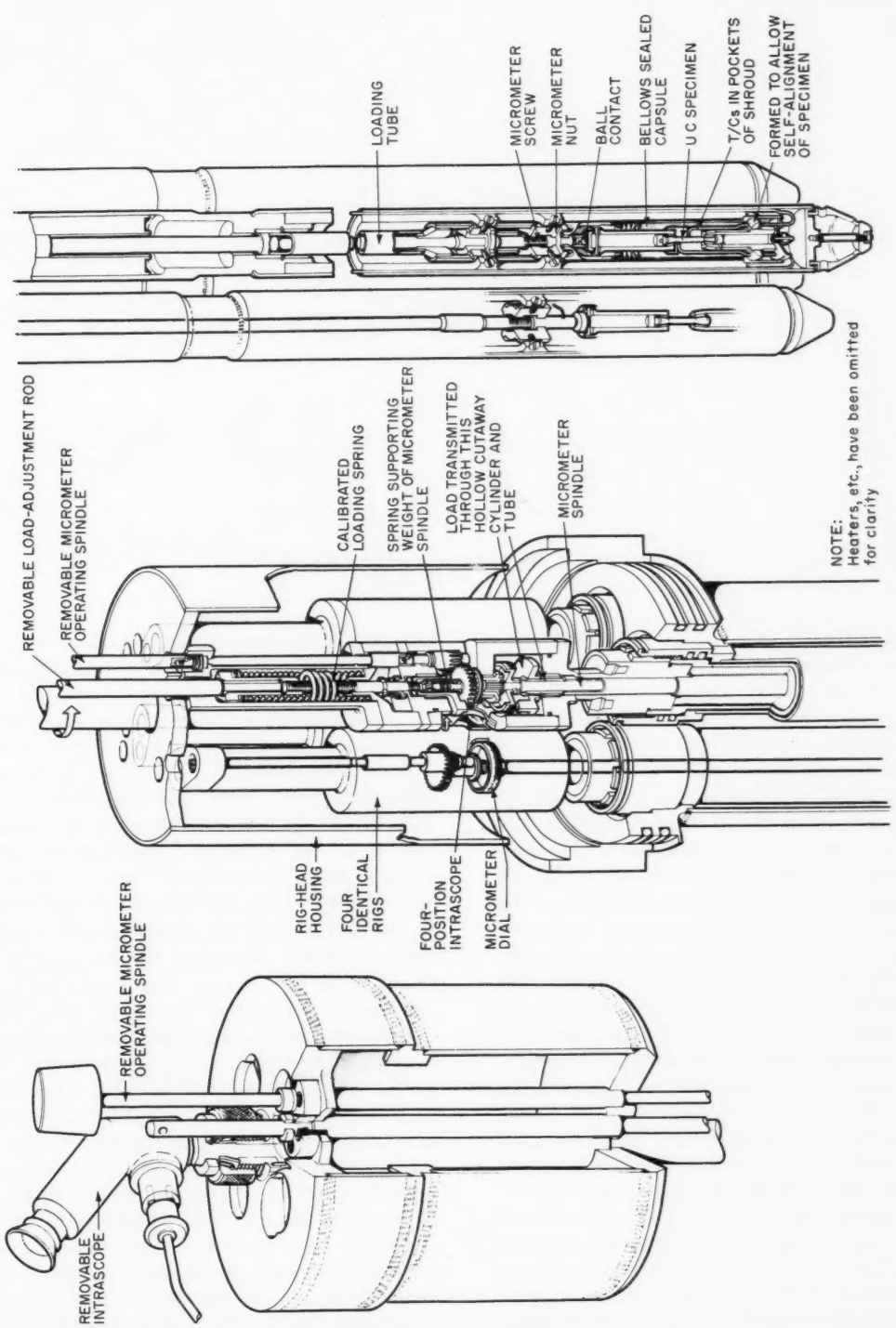

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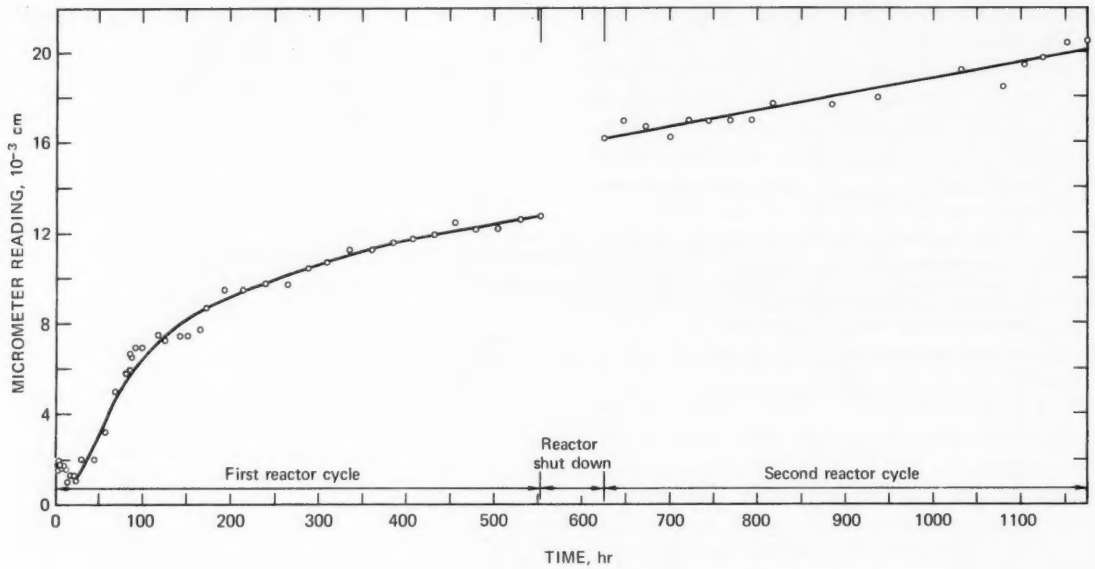

Fig. 11 Creep of UC during irradiation[18] by Harwell at 800°C.

to 1300°C, (2) controlled compressive stress of up to 13,000 psi, (3) initial fission rate of about 2×10^{13} fissions/(cm^3)(sec) in the Battelle Research Reactor with potential extension to higher flux facilities, and (4) the ability to accommodate uranium as well as mixed uranium–plutonium oxide fuel specimens. The developed design is shown schematically in Fig. 12. The annular specimen configuration, nominally 0.25 in. in outside diameter, 0.05 in. thick, and 0.75 in. long, was selected on the bases of fabricability, loading, and heat transfer. The thin annulus minimized transverse fission rate and temperature gradients within the specimen. The specimen was supported between two thoriated tungsten platens by the upper and lower load columns. The load columns, in turn, rest between alumina centering and support members. The upper support was fixed, whereas the lower support moved as the fuel specimen deformed. Load was applied to the specimen through a double-bellows-sealed helium-driven piston that acted on the bottom support member. Specimen stresses of 500 to 13,000 psi could be applied and controlled to ±0.5%. Relative axial motion between the top and bottom surfaces of the specimen was transmitted to a pneumatic displacement transducer by two concentric alumina probes. The pneumatic displacement transducer was of the design developed in Canada and previously described. The strain-measuring system as used at Battelle's Columbus Laboratories was capable of following specimen deformation to ±0.0001 in. over a useful range of 0.090 in.

The temperature of the specimen was controlled by electrical-resistance heaters made up of two concentric alumina tubes surrounding the specimen and its immediate supports, with stranded W–26 Re wire wrapped in fully recessed grooves on the outside of each tube. The inner tube supported the primary specimen heater, and the outer tube held the top and bottom guard heaters. The three heaters were grounded to the capsule wall, with power to each controlled independently. Stranded W–Re thermocouples (W–26 Re vs. W–5 Re) were used to monitor temperatures in the hot zone of the capsule, including the fuel-specimen surface. A junction was formed on the fuel surface by tying a 1-mil tungsten foil to the surface with the thermocouple wires. This arrangement gave a more reliable indication of fuel-surface temperature than did tying the two wires in a knot and then holding the knot against the surface, as had been done in the initial experiments in this program. The temperatures in cooler portions of the capsule were monitored by means of Inconel-sheathed, alumina-insulated, Chromel–Alumel thermocouples. A self-powered neutron detector was used to monitor the thermal-neutron flux in the vicinity of the fuel specimen.

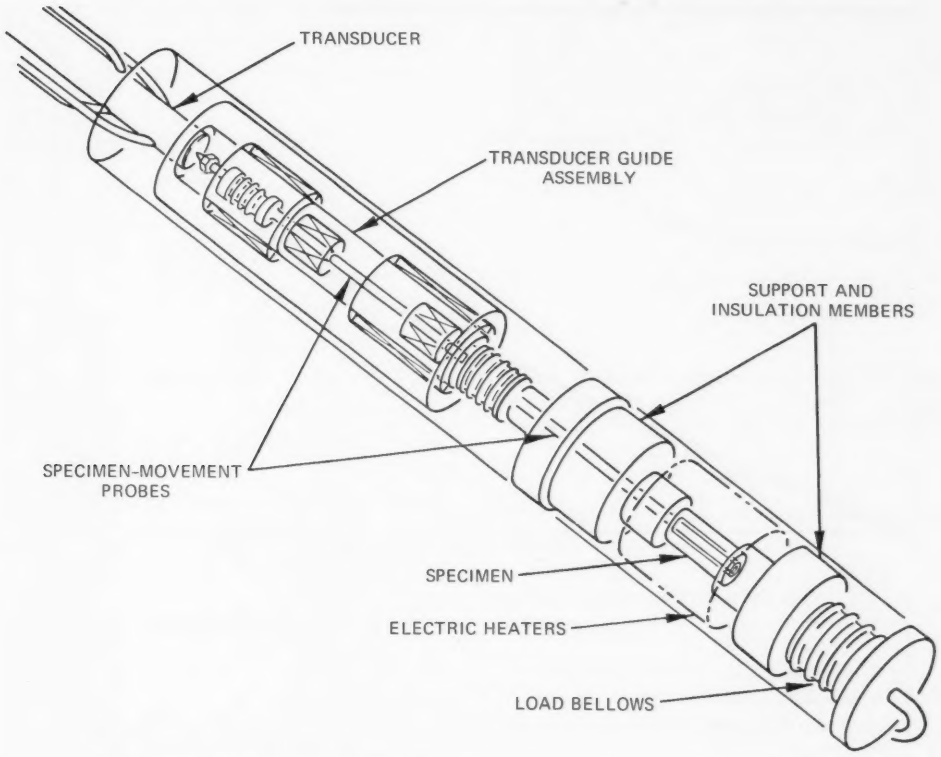

Fig. 12 Schematic of the Battelle Memorial Institute, Columbus Laboratories, in-pile creep apparatus.[20]

A number of successful in-pile creep experiments have been conducted with UO_2 fuel samples using the Columbus Laboratories capsule. Figure 13 shows representative in- and out-of-pile results for a particular set of conditions. The results of the experiments with UO_2 fuel specimens are given in Ref. 21. Experiments with mixed uranium–plutonium oxide fuels are under way.

Creep of Uranium Oxide Helices

Argonne is also making a study of fission-induced creep of ceramic fuels.[22] An experimental capsule has been designed to continuously measure the creep deformation of a UO_2 helix at fission rates up to about 3×10^{13} fissions/(cm^3)(sec). A helical-spring specimen geometry has been selected to overcome many of the experimental difficulties associated with creep measurements on ceramic fuels. Ceramics are susceptible to stress concentrations from the loading configuration and to the presence of microcracks that are inherent in such materials. Under uniaxial elongation or compression of a helical specimen, the stress conditions are largely torsional, which is a favorable stress mode for the evaluation of mechanical properties. Also, the spiral geometry of the spring offers a practical magnification of several orders of magnitude in the creep measurements, i.e., for the geometries considered, a creep rate of 10^{-7} hr^{-1} would result in a spring deflection rate of about 10^{-4} cm/hr. This reduces both the time and the precision required for the experimental measurements.

A substantial effort in this program has been devoted to the fabrication of UO_2 helices of the desired geometry and sufficient quality for in-reactor experiments.[23,24] A slip-casting technique was developed which could produce UO_2 helices with up to 27 turns, outside diameters of 22.2 to 23.8 mm, densities 94.0 to 96.5% of theoretical, and a circular wire cross section of about 1.8 mm.

The experimental capsule was designed for a 5- to 20-turn helix with nominal dimensions of 25 mm in outside diameter and 1.75 mm in wire diameter.[25] The specimen was supported by two zirconium grips. The

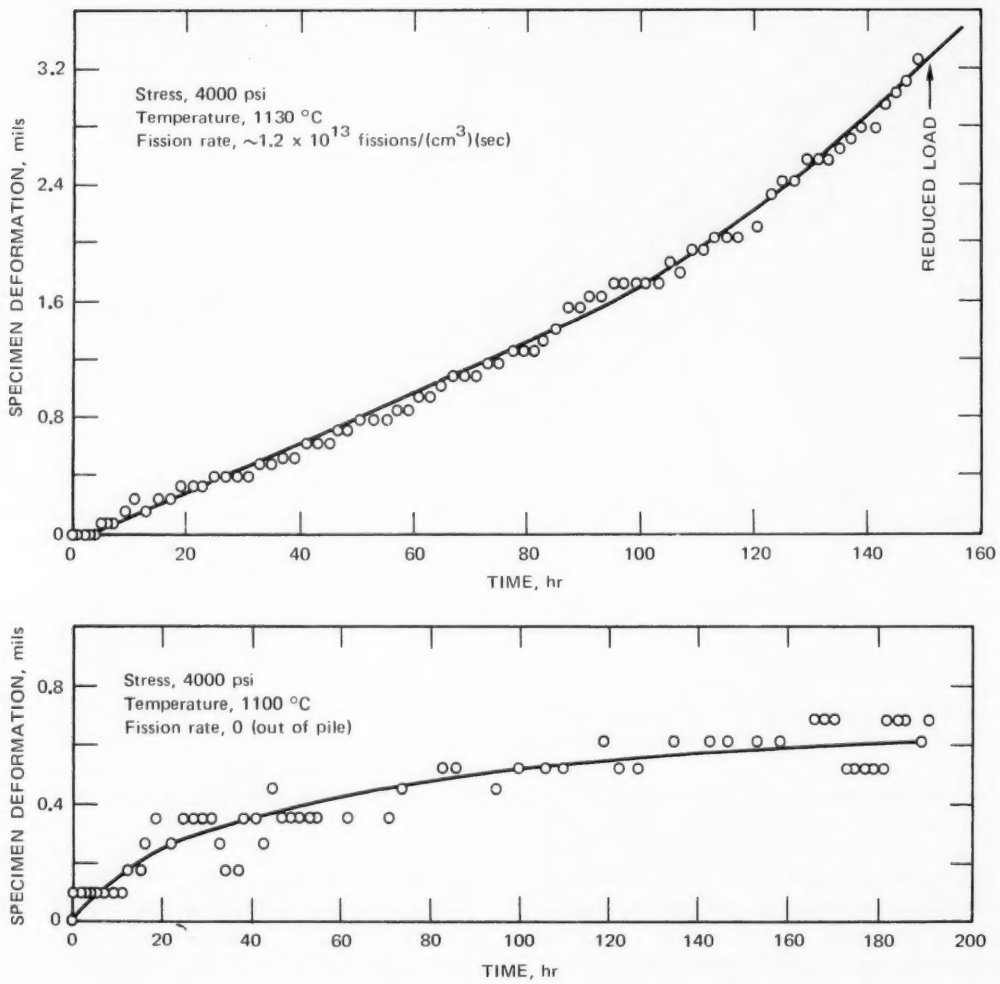

Fig. 13 In-pile and out-of-pile deformation of UO_2 specimens under compression as a function of time, observations at Battelle Memorial Institute, Columbus Laboratories.[21]

upper grip was fixed, and the lower grip was supported by a bellows and spring assembly. Load on the specimen was controlled by varying the pressure acting on the bellows and also by changing the deadweight that acted on the bellows. Figure 14 shows the specimen supports and the loading bellows. A radiation-resistant LVDT at the top of the capsule was used to monitor the deflection of the specimen through a tubular pushrod that was supported by the lower specimen grip. A control sample identical to the creep specimen was included in the capsule to provide a comparison of fission effects in the helical specimen under conditions of no external load. The lower half of the capsule was filled with NaK for improved heat transfer. Continuous temperature monitoring was provided in the vicinity of the LVDT, control sample, and creep specimen. No temperature control was included in these experiments because of the extremely low test temperatures, where thermal-creep effects are expected to be insignificant. Self-powered neutron detectors were incorporated in the capsule to provide a continuous measure of the flux level during the irradiation.

Figure 15 shows the results of the initial experiment in terms of the maximum shear-strain rate as a function of maximum shear strain.[26] The irradiation took place in the V-RAFT facility of the General

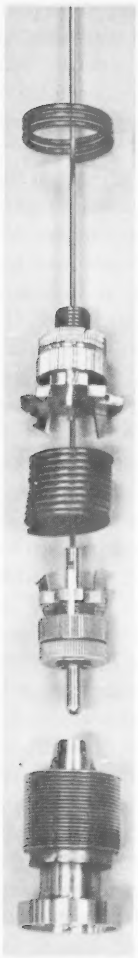

Fig. 14 Helical UO_2 specimen, supports, and loading bellows for ANL instrumented creep capsule.[25]

Electric Test Reactor. The creep specimen was under a maximum shear stress of 2890 psi at a surface temperature of 100°C and a fission rate of 2.6×10^{12} fissions/(cm^3)(sec). Extrapolation of these data is reported to result in an estimated steady-state shear-strain rate of 6×10^{-6} hr^{-1}. The specimen in the first in-pile experiment in this program fractured as a result of an erratic capsule movement during a relocation operation.

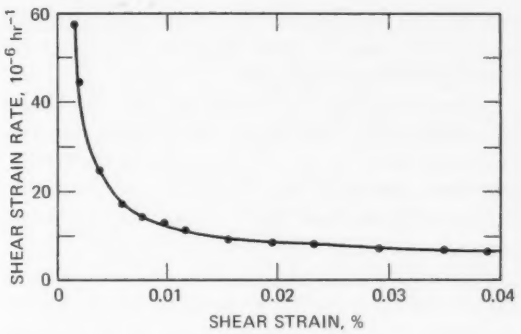

Fig. 15 Relation[26] between in-pile shear-strain rate and shear strain for a UO_2 helix tested by ANL under a maximum shear stress of 2890 psi at 100°C and 2.6×10^{12} fissions/(cm^3)(sec).

In-Pile Mechanical Properties of Ceramic Fuels

The viscosity and yield strength for four different nuclear fuel materials in a reactor environment were measured by Knolls Atomic Power Laboratory.[27] The four materials were UO_2, 2 wt.% UO_2 in ThO_2, 10 wt.% UO_2 in ThO_2, and 39 wt.% UO_2 in ZrO_2; the tests were conducted in the Engineering Test Reactor at temperatures below 750°C and fission rates between 4.0 and 7.6×10^{13} fissions/(cm^3)(sec).

The experiments consisted of compression tests at a strain rate of 1×10^{-5} hr^{-1} and relaxation tests at a constant strain. The fuel, in the form of a 2.38-in.-long annulus with a 0.05-in. wall, was fabricated by isostatic pressing, sintering, and grinding. The experimenters completed the test specimens by sliding inner and outer 0.005-in.-thick Zircaloy-4 claddings over the fuel, adding two 0.500-in. Zircaloy-4 end plugs, and electron-beam-welding the closures. A duplicate, unloaded specimen included in all experiments accounted for effects other than those resulting from the applied load.

The test equipment consisted of a strain-controlled tensile-test apparatus designed to fit in a standard in-pile loop modified to apply a compressive load to the specimens. The motor and drive mechanism for the system were outside the reactor. The specimen load was provided by the action of the in-pile loop pressure against a piston that was attached to a pull-rod tension member. The motor positioned a movable stop connected to the piston and could also augment the load

provided by the piston. The motor was a stepping motor that rotates a fixed distance (0.233 μin.) for each applied electrical impulse.

The deformation of the specimen was determined by a comparison of the known displacement caused by the motor with the total displacement of the system. The latter consists of the deformation of the system due to the applied load, which had been determined by previous calibration, and the deformation of the specimen, which is sought. This method measured specimen strains to within about 1 mil. The mechanical properties of the fuel materials were determined from the strain data by an analysis of the entire system, including the fuel, cladding, and loading members.

Specimen temperatures and fission rates were calculated on the basis of the individual ^{235}U loading, measured neutron flux, and measured loop-water exit temperature.

DISCUSSION

This article has covered the principal experimental studies on in-pile creep of reactor materials; it is by no means exhaustive. The in-pile creep equipment developed at Grenoble has been described by Cartier.[28] The development of differential transformers for measurement of in-pile fuel elongation has been successfully undertaken at Halden.[29] Will[30] has reviewed a number of in-pile strain-measuring techniques that have been proposed and/or utilized by others.

A rather broad range of experiments for the study of the effect of radiation on the creep of reactor materials has been encountered in this article. These in-pile experiments have ranged from simple dead-weight-loaded specimens without any instrumentation to extremely elaborate capsules that provide continuous measurement and control of the specimen loading as well as its environment. The simple, noninstrumented experiments appear to have the obvious advantages of ease of fabrication, low cost, minimum of surveillance, and ability to accommodate a number of specimens in a single experiment. However, the lack of definitive information on the conditions, particularly temperatures, seen by the specimens can detract greatly from the value of the results. In some instances, in fact, the results from noninstrumented in-pile creep experiments have been completely discarded because the uncertainties resulting from the lack of specimen-temperature data made the results meaningless.

Precise knowledge and control of specimen temperature during the irradiation appear to be the essential requirements for meaningful in-pile creep experiments. For irradiations with prepressurized tube specimens, consideration must be given to the change in specimen stress as the tubes distend during the course of the experiment, the change in the stress level being closely related to the initial gas volume and the amount of strain encountered. In noninstrumented experiments, specimen strain is determined from post-irradiation examinations and/or interim measurements during pile shutdowns. Such discrete and generally infrequent observations yield little or no information on transient creep characteristics, such as occur during early stages of the experiment.

The incorporation of temperature-monitoring and -control capability will greatly enhance the value of an in-pile creep experiment without overly complicating the design. For example, the design and fabrication of multiple-specimen capsules with temperature-control capability are considered to be relatively straightforward.[18] Even the maintenance of constant specimen stress does not appear to be overly difficult; e.g., loading by deadweights is the simplest, and the maintenance of a desired pressure in a sealed specimen does not greatly complicate the picture. It is, then, the addition of a continuous in-pile strain-measuring capability that appears to be the most complex aspect of instrumented in-pile creep experiment design. This capability generally limits a given experimental rig to the use of a single specimen. It is interesting to note that several approaches have been used successfully for in-reactor strain measurement. All these approaches have resulted in highly complex experiments with stringent design, fabrication, and operational requirements.

In the area of temperature control, it is worthwhile to note that control of the temperature of the experiment as a whole and not only the specimen itself must be considered. Although maintenance of a constant specimen temperature is required if interpretable creep behavior is to be obtained, control of the temperatures of capsule components is often mandatory for meaningful measurements of specimen behavior. Such considerations as temperature sensitivity of LVDTs, thermal expansion of various components, localized overheating, etc., have been previously noted.

Since the operating conditions can change in the facility in which the experiment is conducted, spurious indications of substantial magnitude can be introduced, as is excellently illustrated in Fig. 11. For this reason, isolation of the experiment from the reactor environment, as well as some control of the temperature of the entire experiment, should be considered in the design.

A variety of specimen designs has been utilized in the programs considered. For experiments with the common cladding and structural materials, specimen design (geometry) is obviously no problem. However, with ceramic fuel materials, specimen design is a much more significant consideration. Here again a variety of configurations, ranging from simple rectangular bars to helical springs, was encountered. Special development was obviously required for the fabrication of the latter specimens. As would be expected, specimen misalignment, thermal shock, and mechanical jarring often led to catastrophic results in the studies with ceramic fuel materials.

Chromel–Alumel thermocouples with mineral insulation and either stainless-steel or Inconel sheaths are the most common devices for temperature measurement, although more exotic combinations have also been used. Heater design is generally similar, typically consisting of Nichrome resistance wire insulated from Inconel or stainless-steel sheaths (or supports) by magnesia or alumina; again, other combinations have been used.

A number of successful approaches to the in-pile measurement of creep of reactor materials have been reviewed in this article. Any particular approach must necessarily be determined by the specific objectives of the particular experiment and the restraints imposed by the reactor facility to be used. It is doubtful that any one approach is superior to the others. The most meaningful experiments incorporate measurement and control of specimen temperature, as well as continuous measurement of specimen strain and maintenance of constant stress. The achievement of these capabilities for in-pile experiments leads to extremely complex experiment designs and stringent operating requirements. This complexity necessarily limits the number of experiments that can be conducted and hence also limits the rate at which the desired information on the effect of radiation on the creep of materials is developed.

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I wish to thank A. B. G. Washington of the Dounreay Experimental Reactor Establishment, D. J. Clough of the Atomic Energy Research Establishment at Harwell, and C. D. Williams of Atomic Energy of Canada Limited for providing material used in the preparation of this article. Thanks are also expressed to my colleagues at Battelle Memorial Institute's Columbus Laboratories, particularly E. M. Simons, for their assistance.

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A variety of specimen designs has been utilized in the programs considered. For experiments with the common cladding and structural materials, specimen design (geometry) is obviously no problem. However, with ceramic fuel materials, specimen design is a much more significant consideration. Here again a variety of configurations, ranging from simple rectangular bars to helical springs, was encountered. Special development was obviously required for the fabrication of the latter specimens. As would be expected, specimen misalignment, thermal shock, and mechanical jarring often led to catastrophic results in the studies with ceramic fuel materials.

Chromel–Alumel thermocouples with mineral insulation and either stainless-steel or Inconel sheaths are the most common devices for temperature measurement, although more exotic combinations have also been used. Heater design is generally similar, typically consisting of Nichrome resistance wire insulated from Inconel or stainless-steel sheaths (or supports) by magnesia or alumina; again, other combinations have been used.

A number of successful approaches to the in-pile measurement of creep of reactor materials have been reviewed in this article. Any particular approach must necessarily be determined by the specific objectives of the particular experiment and the restraints imposed by the reactor facility to be used. It is doubtful that any one approach is superior to the others. The most meaningful experiments incorporate measurement and control of specimen temperature, as well as continuous measurement of specimen strain and maintenance of constant stress. The achievement of these capabilities for in-pile experiments leads to extremely complex experiment designs and stringent operating requirements. This complexity necessarily limits the number of experiments that can be conducted and hence also limits the rate at which the desired information on the effect of radiation on the creep of materials is developed.

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Tritium in Molten-Salt Reactors

By R. B. Briggs*

Abstract: *Molten-salt breeder reactors are expected to generate tritium at a rate of about 1 Ci/MWd(t). Neutron reactions with lithium, an essential ingredient of a breeder fuel salt, are the major source of the tritium. At 800 to 1300°F, the operating temperature range of molten-salt reactors, tritium tends to diffuse through the metal walls of the reactor equipment and piping into the surroundings. Measurements of the tritium distribution in the Molten-Salt Reactor Experiment (MSRE) indicated that about 20% of the tritium escaped from the reactor systems in this way. A method was developed for calculating the distribution of tritium in the MSRE, and reasonable agreement with the measured distribution was obtained by adjusting parameters in the equations. When the same methods were used to calculate the tritium distribution in a large molten-salt breeder reactor, the calculations indicated that most of the tritium might diffuse into the steam-power system of current plant designs. Several modifications in plant design, or in operating conditions, have the potential for confining substantially all the tritium to the reactor systems and containment cells. Research is in progress to provide the data necessary to design a plant for suitable containment and disposal of the tritium.*

The fuel salt proposed for use in molten-salt breeder reactors (MSBRs) contains uranium tetrafluoride and thorium tetrafluoride dissolved in a lithium fluoride–beryllium fluoride carrier salt.[1] A typical composition, expressed in mole % of each constituent, is 71.6 LiF–16 BeF_2–12 ThF_4–0.4 UF_4. The lithium fluoride–beryllium fluoride based salts, in which the lithium is 99.99+% ^{7}Li, are the only ones known to satisfy the requirements of a fuel for a breeder reactor. These requirements include readily available constituents, liquidus temperature below 1000°F, good fluid flow and heat-transport properties, and low cross section for parasitic absorption of neutrons. One adverse characteristic is that tritium is produced at high rates in such fuel salts. Other radioactive materials are generated in much larger quantities, but the tritium is more difficult to contain. It can diffuse through metal at the high temperature of MSBRs,[2] and this requires special consideration in their design and operation.

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TRITIUM PRODUCTION IN MSBR

In an MSBR plant of 1000 MW(e) generating capacity, the reactor would have a thermal output of 2250 MW and would produce tritium at a rate of about 2420 Ci/full-power day.* The sources[3] of this production are shown in Table 1. Although ^{6}Li is consumed in producing tritium, the concentration in the salt is maintained by two mechanisms. About half the ^{6}Li is replaced by the neutron reaction $^9Be(n,\alpha)^6Li$ with the beryllium in the salt. The remainder is supplied by the addition of new salt to replace salt that is discarded in fuel reprocessing.

Table 1 Sources and Rates of Production of Tritium in a 1000-MW(e) MSBR

	Production rate, Ci/day
Ternary fission	31
$^6Li(n,\alpha)^3H$	1210
$^7Li(n,\alpha)^3H$	1170
$^{19}F(n,^{17}O)^3H$	9
Total	2420

*By comparison, the average rate of production of tritium in plants of 1000 MW(e) capacity is estimated to be 40 to 50 Ci/day for light-water, high-temperature gas-cooled, and fast breeder reactors and 3500 and 5800 Ci/day for heavy-water reactors.

CHARACTERISTICS OF AN MSBR PLANT

Present concepts of molten-salt reactor power plants[4] can be represented by the arrangement of equipment and cells shown in Fig. 1. In the primary system, fuel salt is recirculated through the reactor vessel, where it is heated by fissioning of the uranium as the salt passes through the graphite core, and through heat exchangers where it is cooled. Means are provided for contacting the salt with an inert purge gas, such as helium, in order to transfer krypton, xenon, and tritium to an off-gas system. In the secondary system a coolant salt is recirculated through the primary heat exchangers and through steam generators–superheaters. A purge-gas system is provided also, primarily to protect the salt from air and moisture. Steam from the steam generators–superheaters is expanded through a turbine generator to produce electricity. Treated condensate is returned to the steam generators.

The primary and secondary systems are installed in shielded cells to protect the operators from radiation emitted by the reactor and the process fluids. The equipment must be heated to 1000°F to keep the salts molten. This is accomplished by building the cells as furnaces or by applying heaters and insulation to the piping and vessels. The cells must be designed to contain the radioactive or chemically toxic liquids and gases that would be discharged from breaks in the piping and would be adequate to confine any tritium that passed from the equipment into the cells. The amount of radioactivity in the turbine-generator plant should be kept small enough so that the equipment can be installed in a well-ventilated industrial building, maintenance can be accomplished with few special precautions, and water, steam, and noncondensable gases can be discharged safely to the surroundings.

MEASURED DISTRIBUTION OF TRITIUM IN MSRE

Some experience[5] with tritium in a molten-salt reactor was obtained during the operation of the Molten-Salt Reactor Experiment (MSRE) from 1965 through 1969. Disposal of the tritium from that reactor was never a notable problem, and for the first several years the only measurements were those made for health-physics monitoring of liquid wastes. In 1969 increasing awareness of the importance of tritium in large molten-salt reactors prompted measurements of

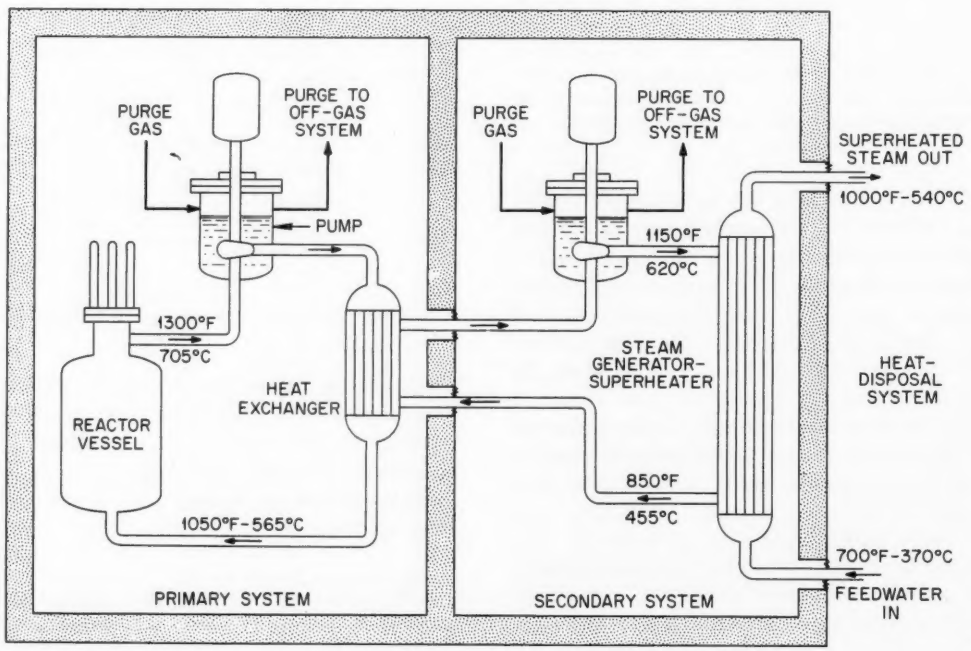

Fig. 1 Diagram of a molten-salt reactor plant.

the distribution of tritium in the MSRE for comparison with the calculated production and with calculations of the distribution.

Figure 1 is reasonably representative of the MSRE configuration. The reactor operated at a full power of about 7.3 MW(t) and had one primary and one secondary heat-removal circuit and a maximum temperature of 1210°F. The fuel salt was similar to the MSBR fuel salt. A radiator was used in place of a steam generator–superheater. Air was blown over the radiator tubes and discharged up a stack to the atmosphere. Fuel salt was sprayed into the helium cover gas above the free surface of salt in the tank of the primary system pump on an 8-min cycle. A purge of cover gas was passed from the pump tank through pipelines and a large bed of activated carbon and was then discharged up a stack to the atmosphere. In the secondary system there was a small bypass of salt through the pump tank, and the purge gas was passed through a small bed of carbon before being discharged to the atmosphere.

The tritium generation rate in the MSRE during the time that the distribution was being measured was calculated to be about 54 Ci/full-power day in the fuel salt. A "best estimate" of the distribution of the tritium obtained from analysis of the measurements is shown as the measured distribution in Table 2. Best estimate is used to describe the distribution because some of the very high and very low measurements were not included in the analysis. Also, the samples of the off-gas were taken downstream of the carbon beds, and the analysis had to take into account the continuing release of tritium for long periods after the power was shut off. One set of several self-consistent measurements of the amount of tritium in the cooling air from the radiator was not used in arriving at the distribution. Those measurements indicated that only about 0.4 Ci/day of tritium passed through the radiator tubes into the air. The possibility remains that this much lower rate is the correct one.

Table 2 Comparison of Measured and Calculated Distributions of Tritium in the MSRE

	Removal rate, Ci/full-power day			
		Calculated		
Location	Measured	Reference case	Range of cases	Best fit*
Graphite in core	8	16	7 to 40	11
Primary-system off-gas	25 to 30	24	12 to 42	34
Secondary-system off-gas	0.6	0.06	0.0 to 0.6	0.2
Air from radiator	3 to 5	4	0.8 to 6	3
Reactor cell	3 to 5	9	1 to 11	5
Coolant cell		1	0.2 to 1	0.8
Unaccounted for	14 to 5			

*Mass-transfer coefficients = 0.5 x reference case and permeation coefficients for metal = 0.001 x reference case.

CALCULATION OF DISTRIBUTION OF TRITIUM IN MSRE

Tritium is present in the fuel salt as tritium and tritium fluoride in relative amounts that depend on the reducing power (the ratio of UF_3 to UF_4) of the salt. These materials diffuse to metal and graphite surfaces and to liquid–gas interfaces according to the relation[6]

$$\frac{Q}{A} = k\,(C_L - C_S) \tag{1}$$

where, in consistent units,

Q/A = transport rate per unit area

k = mass-transfer coefficient

C_L, C_S = concentrations in bulk salt and at the surface

At metal surfaces, tritium molecules dissociate into atoms and diffuse through the metal at a rate[7]

$$\frac{Q}{A} = \frac{P}{t^n}\left(p_s^m - p_o^m\right) \tag{2}$$

where P = permeation coefficient of metal
p_s, p_o = partial pressure of tritium at inner and outer surfaces of metal
t = metal thickness
$m = \frac{1}{2}$ at pressures >0.01 atm but may increase to 1 at very low pressure
$n = 1$ if $m = \frac{1}{2}$; 0 if $m = 1$

Tritium fluoride can react with the metal to release tritium atoms, but this reaction is assumed to be relatively slow and a negligible source of tritium atoms in the metal.

On reaching graphite surfaces in the reactor core, tritium and tritium fluoride diffuse into the pores and are sorbed on surfaces. Because the amount found on graphite removed from the MSRE was so large, we base our calculations on retention by the graphite of all the tritium and tritium fluoride that reaches the surfaces.

Equations 1 and 2 are coupled, and release of tritium and tritutium fluoride to the purge gas is calculated by use of the equation

$$p = HC \qquad (3)$$

where H is the Henry's law coefficient for the solubility of gas in salt.

COMPARISON OF MEASURED AND CALCULATED DISTRIBUTIONS IN THE MSRE

Results of the calculations of the tritium distribution in the MSRE are summarized and compared with the measured distribution in Table 2. Included in the comparison are (1) a reference case based on what were judged to be the most likely values for the system parameters; (2) the range of results obtained by varying the reducing power of the salt, the mass-transfer coefficients, and the permeation coefficient of the metal; and (3) the calculated distribution that was in best overall agreement with the measured distribution.

Considering all the uncertainties, the distribution of the reference case agrees surprisingly well with the measured distribution. However, in the reference case, the amounts of tritium sorbed on the graphite and entering the reactor cell are notably greater and the amount leaving the secondary system off-gas is notably smaller than in the measured distribution. Although the numbers under the range of cases suggest that almost any desired distribution could be calculated by adjusting the values assigned to the parameters, such adjustments cannot be made with complete freedom and still conform to the physical situation. The best fit to the measurements was obtained by simply reducing the mass-transfer coefficients by a factor of 2 and the permeation coefficients for metal in contact with air by a factor of 1000 over the values used in the reference case. The uncertainty in the estimated values of the mass-transfer coefficients is at least a factor of 2. Two effects could cause the effective permeability of the metal to be substantially less than the reference values. The outer surfaces of the piping, the vessels, and the radiator tubes were coated with oxide; oxide coatings can substantially reduce the permeability of metals to hydrogen.[8-11] A second possibility is that the flow of tritium through the metal varies with the first power of the pressure at the very low tritium pressure in the MSRE rather than with the one-half power that was used in the calculations. Use of the first-power relation in the calculations would have the effect of decreasing the permeability by a factor of 100 to 1000.

TRITIUM DISTRIBUTION IN A 1000-MW(e) MSBR

Since a distribution in reasonable agreement with the measurements could be calculated for the MSRE, the same methods with appropriate values for the constants were applied to the Oak Ridge National Laboratory reference design of an MSBR of 1000 MW(e) capacity.[3,4,12] In this design the fuel salt is processed for the removal of krypton, xenon, and tritium on a much shorter cycle than in the MSRE. The purge gas is injected continuously into the circulating primary salt as 0.5-mm-diameter bubbles, and the bubbles with their burden of radioactive gases are removed by centrifugal separators after several passes through the system. The same arrangement can be provided for removing tritium from the secondary salt.

Results of the calculations for the reference-design conditions are shown in Table 3. Also shown are the distributions obtained by reducing the permeability of metal in contact with oxidizing media such as air, water, and steam by a factor of 1000 as seemed to fit the MSRE data best and by several relatively minor changes in the design or in the operating conditions. Graphite was included as an absorber of tritium and tritium fluoride in only part of the cases on the basis that it might saturate and become ineffective. In the distribution, the path of most concern is the flow into the steam system. The other paths can be processed and the tritium can be isolated; but, once tritium enters the steam system, it becomes part of the water,

Table 3 Calculated Distributions of Tritium in an MSBR

	Removal rate, Ci/full-power day				
	Graphite in core	Primary-system off-gas	Secondary-system off-gas	Reactor and coolant-system cells	Steam system
Reference-design conditions: $UF_3/UF_4 = 0.01$	1170	140	30	290	790
		270	50	560	1540
Permeability reduced by 1000*	1350	190	480	130	270
		440	1320	220	440
Permeability reduced by 1000; $H_2 = 10^4\ T_2$†		610	1800	3	5
$H_2 = 10^5\ T_2$		820	1600	<2	<2‡
Reference-design conditions: $UF_3/UF_4 = 0.001$	2310	50	2	20	40
		470	50	500	1400
Permeability reduced by 1000	2320	50	7	10	30
		660	1150	200	410
Side-stream contacting: § $UF_3/UF_4 = 0.01$	460	1760	20	50	130
		2120	40	70	190
Permeability reduced by 1000; $H_2 = 10^4\ T_2$		1220	1200	<2	<2
All permeabilities reduced by 100;¶ $H_2 = 10^4\ T_2$		2330	80	5	5
$H_2 = 10^5 T_2$		2390	30	<2	<2
Side-stream contacting: $UF_3/UF_4 = 0.001$	490	1930	<2	<2	<2
		2420	<2	<2	<2

*Permeability of all metal except the tubes in the primary heat exchangers was reduced by factor of 1000.

†Hydrogen was added to fuel salt at rate of 10^4 times the rate of production of tritium.

‡2 Ci/day is the limit of precision of the calculations; <2 Ci/day indicates that the calculated value was 0.

§Tritium and tritium fluoride are removed from 15% of the primary- and secondary-system flows in side-stream contactors.

¶Permeability of all metal including the heat-exchanger tubes was reduced by a factor of 100.

and the entire steam system, including the turbine, must be contained, or the tritiated water must be discharged to the environment.

The calculations for the reference conditions indicated that as much as 1540 Ci/day (63% of the production) might pass from the primary system through the secondary system into the steam system. Under the more favorable conditions with sorption by graphite and low metal permeability, the rate of transport would be 270 Ci/day. The large flow of tritium into the steam system is simply a reflection of the large heat-transfer surfaces in the MSBR and the small capacity of the purge gas for carrying tritium into the off-gas system.

CONSEQUENCES OF LARGE FLOW OF TRITIUM INTO THE STEAM SYSTEM

If tritium were to enter the steam system at a rate of 1540 Ci/day and the blowdown and usage of water were at the normal rate for a large power plant, the concentration of tritium in the steam and condensate would level out at about 8 μCi/ml of water. This is 80 times the maximum permissible concentration (MPC) in water for occupational exposure.[13] Precautions, such as continuous monitoring, restrictions on handling the water, careful control of leakage, and use of plastic suits and masks for some maintenance procedures, would have to be taken for the protection of plant personnel.

Virtually all the tritium would leave the steam system in the blowdown and would be discharged from the plant in the condenser cooling water. With a flow of 400,000 gal/min in a typical once-through cooling cycle, the tritium concentration in the water leaving the discharge canal would be 0.7×10^{-3} μCi/ml. This is about $\frac{1}{4}$ the current MPC value for water discharged to an uncontrolled area,[13] but 140 times the 5×10^{-3} μCi/liter proposed as the design objective for effluents from light-water-cooled nuclear power reactors.[14] Similar relations would be obtained if the tritium were discharged to the atmosphere through the cooling tower of a closed-cycle-condenser cooling-water system.

The situation is improved if the flow of tritium into the steam and then to the environment is the lower value of 270 Ci/day, but it is clearly desirable, and likely to be required, that the concentration of

tritium in the effluents from a molten-salt reactor plant be at least as low as the proposed design objective for light-water-cooled plants. This would require limiting the discharge from the steam system to about 10 Ci/day.

The release of tritium from the plant might be lowered by reduction of the blowdown and leakage and containment of the tritium in the steam system. At a rate of 1540 Ci/day, the inventory of tritium in the steam system would build up to about 5×10^6 Ci in 12 years and to 10×10^6 Ci by the end of plant life and a concentration of 4 to 9×10^{-3} Ci/ml in the 300,000-gal inventory of water in a typical plant. This concentration is about the same as in the moderator of a large heavy-water reactor,[15] so the same safety precautions would be required. Losses would have to be reduced to 1 to 2 liters per day from the large installation of piping and equipment, much of which operates at pressures above 3500 psi. Attempting to confine the tritium to the steam system does not appear to be a desirable or a practical method of keeping the tritium release to the environment below 10 Ci/day.

METHODS OF REDUCING THE ESCAPE OF TRITIUM INTO THE STEAM SYSTEM

Several modifications in the design or operation of the reference plant, separately or in combination, have the potential for drastically reducing the amount of tritium that escapes into the steam system and in some instances into the reactor and coolant-system cells. These modifications involve adding hydrogen to the primary salt, reducing the permeability of the metal walls, substituting side-stream contacting of salt and gas for injection of gas bubbles into the primary and secondary systems, exchanging tritium for hydrogen in hydrogenous compounds, and using other fluids to couple the primary system to the steam generators. Several possibilities under consideration are discussed in the following paragraphs.

We have inferred from the MSRE experience that the effective permeability of the metal may be lower than our reference values by a factor of 1000. The numbers in Table 3 show that this effect alone is not sufficient to reduce the tritium reaching the steam system to 10 Ci/day or less. If, however, hydrogen is added to the fuel salt in amounts of 10^4 and 10^5 times the rate of production of tritium, then the corresponding calculated flows of tritium into the steam system are 5 Ci/day and <2 Ci/day (the limit of precision of the calculation), respectively. For the addition of hydrogen to be effective, the reduced permeability must result from a barrier through which the flow of hydrogen is proportional to the square root of the pressure and not to an effect that causes the exponent of the pressure to increase to a value of 1 at low pressure.

The rate of removal of tritium and tritium fluoride from the primary and secondary salts can be increased by contacting side streams of the salts with larger flows of purge gas in packed columns, spray towers, or other types of contactors. With side-stream contacting of 10,000 gal/min each of primary salt (15% of the flow in the primary system) and secondary salt, addition of hydrogen is shown in Table 3 to be effective with a 100-fold reduction in the permeability of the metal. Tungsten is compatible with fluoride salts, and the 100-fold reduction in permeability and a permeation rate proportional to the square root of the pressure could be obtained with a sound 0.1-mm coating on the interior surfaces of the primary and secondary systems. Coating the entire reactor systems might be impractical, but the amount of tritium reaching the steam system could be reduced to 5 Ci/day for $H_2/T_2 = 10^5$ by coating only the tubes in the primary heat exchangers and in the steam generators.

Side-stream contacting makes possible the use of larger flows of purge gas than does the bubble-injection system and the stripping of tritium fluoride to lower concentrations in the primary salt. If the fraction of tritium present as the fluoride is increased by reducing the concentration ratio UF_3/UF_4 in the primary salt from 0.01 to 0.001, practically all the tritium can be discharged into the primary-system off-gas as tritium fluoride. This finding rests on the assumption that the rate of reaction of tritium fluoride with the metal surfaces will be low at the low concentrations of tritium fluoride in the salt.

No other manipulations in the primary system seem likely to have much effect on the tritium distribution, but the secondary system offers several additional possibilities. The sodium fluoroborate coolant salt proposed for use in the secondary system seems to contain small amounts of a hydroxyl compound without being excessively corrosive. Tritium, on entering the fluoroborate, would be expected to exchange with hydrogen in the compound and be retained by the salt. The salt would be processed as necessary to remove the tritium.

The results in Table 3 show that sorption of tritium by the graphite in the reactor core is helpful but not sufficient to prevent the release of excessive amounts to the steam system.

With more drastic changes in plant design, helium containing small amounts of oxygen and water vapor could be used as the coolant in the secondary system. Tritium, on diffusing into the helium, would be oxidized to water and prevented from passing into the steam. Objection to the use of helium in the secondary system is found in the high pressure, the larger primary heat-exchanger surface, and the larger fuel-salt inventory that would be required. These objections might be circumvented by employing the helium in the annuli of dual-wall tubes in the steam generators at the expense of larger and more complicated steam generators.

Use of the nitrate–nitrite salt mixture, generally known as HTS or Hitec, in the secondary system would also keep tritium out of the steam. Tritium entering this salt would be oxidized to water, and the water would be vaporized into the purge gas at high temperature. Thermal instability above 1100°F and potential reactions with graphite if it were to leak into the primary system are objectionable features of this salt. These difficulties could be circumvented by use of the salt in a circulating system between the reactor secondary system and the steam generators.

DISPOSAL OF TRITIUM

After a satisfactory method has been developed for limiting the amount of tritium that enters the steam system, the plant designer is still confronted with removing tritium from the cell atmosphere and the off-gas streams and confining it. This has not yet received much attention. Most of the tritium is likely to be extracted as water or tritium fluoride. The water could be stored in tanks and the tritium fluoride could be sorbed on sodium fluoride beds, or those compounds might be decomposed and the tritium converted to a solid hydride for storage. In any event, excessive dilution by hydrogen must be prevented. A production rate of 2420 Ci/day is equivalent to a trivial 0.8 ml/day of T_2O. The volume of water resulting from a dilution of 10^4 with hydrogen would be of little consequence, but, if the dilution were 10^6, 8000 m^3 of tritiated water would be produced during the life of the plant. Safe storage for such a large volume would be expensive, and means probably would also have to be provided for concentrating the tritium.

RESEARCH AND DEVELOPMENT

Although there is experience with tritium in the MSRE, analysis of its behavior in large molten-salt reactors is largely based on extrapolation of data from the literature and speculation on the chemistry of potential coolants. There is real need for confirmatory experimental data, and a program is in progress to obtain these data. The program includes measurements of (1) the solubility of hydrogen in salts; (2) the permeability of metals and oxide coatings at low partial pressures of hydrogen; (3) the capacities of graphite and of potential coolants to retain hydrogen and tritium under simulated reactor conditions; and (4) reaction rates of hydrogen fluoride in low concentrations in salts with metals. Investigation of methods for separating tritium compounds from process streams and from cell atmospheres and for storing the tritium safely and economically while it decays will be included later.

CONCLUSIONS

Experience with the MSRE and calculations indicate that the high generation rate of tritium and its ability to diffuse through metal at high temperature requires special attention in the design of large molten-salt reactors. The present major concern is to limit the amount of tritium that passes into the steam in the turbine-generator system. Several methods offer promise for adequately controlling the distribution of tritium in the plants. Research and development are being conducted to establish their technical feasibility. Subsequent studies will be required to determine whether incorporation of processes for containing the tritium will have an important effect on the cost of power from a large plant.

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AMERICAN NUCLEAR SOCIETY—CRITICAL REVIEWS

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ANS GUIDELINES FOR PREPARING CRITICAL REVIEWS FOR *REACTOR TECHNOLOGY*

One interpretation of the words "Critical Review" emphasizes the word *critical.* That type of article would have as its purpose the discussion of a single subject, which has been uncertain and perhaps controversial, in considerable depth. An example might be a critical review of the values of alpha for plutonium, assessing all the work done and arriving at a best current estimate. This paper is primarily addressed to specialists on the particular subject, and serves as an authoritative source for the information they use.

In the second interpretation of a Critical Review, the emphasis is on the word *review,* in the sense of survey. Such a paper would be broader and probably more descriptive in scope. A paper on "Solubility of Metallic Elements in Liquid Sodium" is an example of this category. Such a paper would be addressed to a much broader group of readers and written in a fashion that would be of interest to the majority of reactor technologists. It would provide enough of an introduction that the specialist from another field could immediately appreciate why the subject is important.

The criteria for these two types of articles may not be very different:

1. The paper should be based on a thorough coverage of relevant work on the subject from many sources. A review based on the work of only one individual or group is better suited for publication in one of the regular society journals. The review should be critically selective, reporting only the most valid results and indicating why some prior results have a questionable status. The review should call attention to significant gaps where more work is required in the subject of the article.

2. The paper should be timely, on a subject of active current interest.

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4. The paper should be organized so that it is of immediate practical use to the readers. Such organization requires: attention to consistent use of units, presentation of important data in summary curves or tables, and a fully adequate bibliography to the original literature. Details on size, honorarium, style, etc., can be obtained from ANS.

A Review of Reactivity Meters for Operational Fast Breeder Reactors

By Wayne K. Lehto* and C. E. Cohn†

Abstract: *The present techniques for continuous reactivity monitoring are discussed, and their applicability to fast breeder reactors is evaluated. The methods reviewed are those based on the solution of the kinetics equations and those based on measurements of the inherent fluctuations in the mean neutron level. Although no technique is clearly preferable in all applications, meters based on the solution of the kinetics equations appear to be more suitable for reactivity monitoring at power. Because of detector-efficiency limitations, meters based on the noise techniques appear to be impractical for providing a rapid indication of reactivity.*

A brief review of recent work in fast reactor subcritical measurements is also included.

INTRODUCTION

Knowledge of the subcriticality or reactivity of an operational liquid-metal-cooled fast breeder reactor (LMFBR) is an essential prerequisite to safe and economic operation of the facility. For fast reactors having short neutron-generation times, i.e., fast response, it is desirable to have a nearly instantaneous indication of neutron multiplication. Conventional methods of monitoring and ensuring safety of nuclear systems are based on visual observation and electronic surveillance of the outputs of various transducers that monitor the operational parameters of the system. Reactivity in particular is inferred from the variation of the period of the neutron level, as sensed by appropriate neutron detectors and displayed on a period meter. Log and linear displays of the neutron level provide additional control and reactivity indication.

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An alternative to these conventional approaches is monitoring and measuring reactivity changes with reactivity meters, some of which provide the rapid response desirable in fast systems. These instruments sense the variations in neutron level and, after processing the signal electronically, produce an output proportional to reactivity. Two widely divergent techniques provide this indication of reactivity. The first is called the inverse-kinetics or power-history method and has been employed with considerable success in both thermal[1-6] and fast[6-9] reactors in both the shutdown (subcritical) and supercritical conditions and during operation at power. This technique was originally developed and applied to measure reactivities in a heavy-water-moderated critical facility at low power and has not as yet been applied to thermal reactors at power, although considerable at-power experience has been attained in fast systems.[6,7,9]

The second technique, based on the fluctuations of the neutron level about its mean value (noise), is more suitable to the just-critical and subcritical range and, as yet, has not been applied satisfactorily to a power reactor, although studies of this problem have been made.[10-13]

The conditions under which reactivity measurements are made in a reactor at power and in the same reactor at zero power or subcritical are considerably different. At low power or subcritical, the fission-product gamma-ray fields preclude the use of higher efficiency boron or ^{3}He ion chambers. Consequently the only practical detectors are fission counters which, with their lower detection efficiencies, compromise the applicability of some of the reactivity-monitoring techniques. Further problems encountered at low-power operation are the effects of changing neutron

sources that occur during fuel-loading changes and because of plutonium buildup, in particular ^{240}Pu, in the fuel and blanket regions. At high-power operation the problems caused by the fission-product gamma rays do not play a role; in this case ion chambers are suitable and, in fact, desirable because of the burnup and saturation problems associated with the fission chambers.

The accuracy required from a reactivity-monitoring device varies with the degree of subcriticality. At or near criticality, knowledge is required of reactivity changes accurate to several parts in one hundred, whereas, in a full-shutdown fast reactor [such as the Fast Test Reactor (FTR)], errors of ±4 dollars at −30 dollars are acceptable.

This report describes these two techniques of reactivity measurement and assesses their applicability as reactivity-indicating instruments in operational LMFBRs.

Other techniques of reactivity measurement, such as source multiplication, are being considered for shutdown-margin indication in the FTR to protect against errors in fuel loading. These techniques do not satisfy the rapid response requirements of a reactivity meter; however, they are discussed here as alternatives to reactivity meters at low-power operation. The application of these techniques to operational fast breeder reactors is currently under development, and a critical review of their performance is premature at this writing.

REACTIVITY METERS BASED ON THE SOLUTION OF REACTOR-KINETICS EQUATIONS

Direct determination of reactivity is made by solving reactor-kinetics equations using the output of a neutron detector as input to an analog computer or special-purpose hard-wired analog equipment.

The kinetics equations are

$$\frac{dn}{dt} = \left(\frac{\Delta k - \beta}{l}\right) n + \sum_{i}^{j} \lambda_i C_i + S$$

$$\frac{dC_i}{dt} = \frac{\beta_i n k}{l} - \lambda_i C_i, \quad i = 1,2,3, \ldots j$$

where n = instantaneous neutron population

Δk = reactivity

β_i = fraction of delayed neutrons from group i

β = total delayed-neutron fraction

t = time

l = prompt-neutron lifetime

λ_i = ith group decay constant

C_i = concentration of ith group precursors

S = neutron-source strength

Rearranging, we have for the reactivity

$$\Delta k = \frac{1}{n}\left(l\frac{dn}{dt} + l\sum_{1}^{j}\frac{dC_1}{dt} - lS\right)$$

This equation, using the detector signal as input, has been mechanized to various degrees of sophistication and solved by analog techniques. The delayed-neutron groups have been simulated by both passive[3,6] and active networks;[1,9] some investigators[7] have ignored the term involving dn/dt, arguing that its effect is negligible and that the necessary mechanization to include this term introduces system noise in the output. Inclusion of this term, although not essential in the routine applications of reactivity meters, is necessary to adequately describe small and rapidly varying reactivity changes, i.e., those which occur during several multiples of the prompt-neutron lifetime.

The passive and active methods of delayed-neutron simulation have differing advantages and disadvantages. The passive method employs a network of resistors and capacitors, one of each for every delayed group. The main advantage of this approach is that only one operational amplifier is needed to obtain the delayed-neutron contributions. However, the resistors and capacitors in the network must have odd, nonstandard values, and the delayed-neutron parameters thus are difficult to adjust.

The active method, on the other hand, uses a separate operational amplifier for each delayed group. The delayed-neutron parameters are set by potentiometers and therefore are much easier to adjust accurately and to alter if necessary. The major drawback of this method is the need for more operational amplifiers. However, this drawback is much less significant than previously, since operational amplifiers have undergone large reductions in cost, size, and power consumption since the vacuum-tube era when these techniques originated.

Other investigators neglect the source term[9] and are totally justified whenever the meter is used to detect reactivity changes in a reactor at power. This term must be included in applications at or near criticality (low power), especially in plutonium-fueled reactors with a high ^{240}Pu content. Herein lies a

difficulty with this technique and a source of error. Since the source term is unknown, the value of the instrument setting is adjusted by trial and error.[1,3] The reactor is leveled at a slightly subcritical condition, and a negative step-reactivity change is introduced. The source-term potentiometer is adjusted until the meter output is constant. In this manner[3] the source term can be calibrated to better than 10%. This technique is limited by the inability to recognize the difference between a small portion of an exponential and a straight line in the presence of the inherent system noise.[1] In plutonium-fueled reactors with the inherent ^{240}Pu spontaneous-fission neutron source, application of this technique to reactor startup after fuel-loading changes is questionable because the source term changes with fuel burnup, i.e., the calibration of a reactivity meter would be expected to change whenever irradiated fuel is replaced with clean fuel. The possibility exists of using a combination of digital and analog techniques—the digital calculation could be used to determine a new source term after the loading changes.

A limitation of earlier reactivity meters, particularly those with passive networks, was the limited flux range (two decades) over which the instruments could operate accurately. This problem has been overcome in the meter designed for the Experimental Breeder Reactor II (EBR-II) which is currently set to scale the input automatically over four decades.[9] Should the need arise, the unit can be modified easily to cover several more decades. An alternate approach could use rescaling integrators such as those developed at the Risö Laboratory in Denmark.[14] These analog integrators rescale their contents by a fixed factor on command and have been used successfully in reactor simulators but have not been applied to reactivity meters.

Another advantageous approach is to perform the calculations digitally. This has been done very successfully in a program of critical experimentation.[15-17] Here, a digital computer reads the flux at intervals and executes an inverse-kinetics algorithm. The time interval at which flux is read depends, of course, on the rapidity with which the flux changes.

For best precision the neutron-flux signal should be averaged over the entire time between samplings. The use of point sampling for the flux signal is considerably inferior because the scatter would increase and the precision of the results would naturally suffer. Since an inverse-kinetics algorithm can be considered to be an imperfect differentiator, results are quite sensitive to scatter in the flux readings.

In a typical Argonne National Laboratory (ANL) experiment, the flux was integrated over the sampling interval by putting the chamber current-amplifier output into a voltage-to-frequency converter. The pulses from the latter were counted in a scaler that was read into the computer and then reset at the end of each interval. Such a system has been proposed for detection of anomalous kinetic behavior of a fast breeder.[18]

The digital method can also readily handle the problem of an inherent source. In a typical reactivity measurement, one would continue observing the flux for a period of time after the reactivity change being measured is complete. The computer can then iterate the reactivity calculation, searching for the source term that would give the calculated reactivity the maximum degree of constancy in that final time region.

Since the digital techniques involve a finite data-collection and -processing time, they do not satisfy the instantaneous-output requirements of a reactivity meter. The principles of the digital calculation remain the same as in the analog calculation with hard-wired equipment; however, the flexibility of the digital scheme may warrant acceptance of the delay between input and output.

Reactivity measurements based on the above principles have been used successfully for subcritical measurements, such as rod calibration and sample reactivity worths, in thermal and fast critical experiments and at power in fast reactors. The previous lack of adequate gamma-compensated current chambers has precluded the use of these meters in both thermal and fast subcritical power reactors.

A reactivity meter at EBR-II has been in use[9] since July 1969 and has been particularly useful in monitoring reactivity changes during operation at power, ensuring critical condition prior to rod drops, helping to explain spurious scrams, and verifying the negative power coefficient. The meter has been able to measure reactivity changes as small as $\approx$3.4 cents, the limiting factor being noise on the input channel, partly due to the large detector–core separation.

The reactivity meter at the Dounreay Fast Reactor has been used in the 3- to 30-kW range to measure perturbation sample worths of less than 1 cent and at higher powers to measure control-rod worths of >2 dollars and has also been used for reactivity-feedback-coefficient measurements. When the reactor response to a rod drop at power was being measured, more accurate settings of the feedback parameters were obtained from the reactivity trace rather than the power trace.

The anomalous-reactivity detection system being installed at the Enrico Fermi Fast Breeder Reactor uses digital techniques to calculate the actual reactivity from the observed flux by an inverse-kinetics algorithm and compares this to the reactivity calculated from calibrated control-rod positions and known reactivity-feedback coefficients. Any anomaly or discrepancy between actual and expected reactivity may be taken as an indication of unexpected behavior and may be used to alarm or trip. Limitations of this particular system are a precision of 2 cents in anomalous reactivity and a calculation time of 1.2 sec. The reactivity precision could be improved by averaging the flux signal over the entire time between samplings rather than over the response time of the recorder, as was presumably the case.

An on-line reactivity-anomaly detection system has also been proposed for use at the High Flux Isotope Reactor.[19] This system is designed to aid the operator in detecting and preventing reactivity anomalies by continuously calculating the expected reactivity balance using measured reactor data with an appropriate reactivity model. Requirements of the system are detection of anomalies greater than 50 cents at 30-sec intervals. Workers have made off-line tests by analyzing measured reactor transients using rod-calibration curves to calculate the excess reactivity. The model used was able to track the excess reactivity well within the specified limits. Current work is centered on methods of on-line rod-calibration techniques.

Considerable experience has been gained with the reactimeter in use at the French reactor Rapsodie.[6] Operators there have used that reactimeter as a control device and in assisting monitoring of reactivity changes during startup and approach to power.

As a control device the meter has provided easier and more precise operator control of the power level than conventional methods. In a series of control-rod calibrations, agreement within 3% was obtained with rod worths measured by period techniques. The flux sensor employed with the reactimeter in the initial startup of Rapsodie was a fission chamber.

Reactivity meters, i.e., those of the hard-wired type which provide a nearly instantaneous indication of reactivity, can be or have been applied as supplemental scram and protection devices, malfunction-detection analyzers, and excursion monitors. As a protection device the meter would be preset to actuate an alarm when the system reactivity reaches the preset level. As a malfunction analyzer the meter would give a continuous indication of system reactivity which could then be compared (presumably by digital techniques) to the expected reactivity. Discrepancies would indicate a malfunction and activate an alarm or trip. As an excursion monitor the meter output would be monitored continuously, and analysis after the fact might provide a clue to the cause of the incident. Presumably, workers could implement all these functions simultaneously by sampling and processing the meter output to obtain the desired information.

For at-power operation, where the flux sensor is an ex-core ion chamber, the worth and reliability of reactivity meters have been demonstrated. However, for zero power and subcritical operation, additional work is required to demonstrate these techniques of reactivity monitoring in the LMFBRs. The effects of the changing source strengths present calibration problems, as do the lower detection efficiencies of the fission chambers. Applicability of these techniques to subcritical reactivity monitoring in operational LMFBRs depends entirely on the ability to attain adequate detection efficiencies through detector design and/or placement within the core. The gamma background present in an operational reactor further affects the precision of the measurement. A detector system has been developed which can be operated in the hostile gamma environments of fast power reactors.[20] The detector is a ^{235}U (99.9%) fission chamber depleted in ^{234}U to reduce alpha activity. Special design features were incorporated to reduce alpha and gamma pileup. The detector coupled to an improved current-mode preamplifier was capable of operating in a gamma flux of 10^7 R/hr with a loss of only 34% in neutron sensitivity. A similar detector placed in a typical measurement position in the ZPR-9, FTR mockup had a measured efficiency of 2.8×10^{-6} events/fission.[21] Whether efficiencies of this magnitude can be obtained in a larger LMFBR core remains to be seen. In the present LMFBR Demonstration Plant Mockup, ZPPR Assembly 2, at the Argonne National Laboratory in Idaho, routine single measurements of subcritical reactivities down to 10 dollars with a precision of ±10% are made with detectors having efficiencies of $\approx 5 \times 10^{-6}$ events/fission.[22] This error is expected to vary inversely as the square root of the detector efficiency and to decrease as the square root of the measuring time or number of repetitions of the measurement.

Assuming that a reasonable value of detector efficiency can be attained and that the source term and changes in detector efficiency can be accounted for, reactivity meters based on the above principles should prove useful in monitoring reactivity at or near critical in operational LMFBRs.

REACTIVITY METERS BASED ON ANALYSIS OF THE NEUTRON FLUCTUATIONS

Reactivity determinations can be made alternatively if the prompt-neutron decay constants are measured by analysis of the inherent fluctuations (noise) in the neutron level. These fluctuations are initiated by the discrete nature of the fission, capture, and leakage processes, which constitute a white noise input to the reactor. The spectral density (mean square amplitude of the fluctuations per unit frequency) is proportional to the reactor transfer function, which is strongly dependent on the kinetic parameters of the system.[10,11] Reactivity is one of these parameters. If delayed neutrons and feedback effects are neglected, the reactor behaves as a first-order filter with constant spectral density for frequencies much less than the break frequency. Above this the spectral shape is inversely proportional to the frequency squared. In fast systems, delayed neutrons and feedback effects produce frequency components in the spectral density well below the break frequency, which is the reactivity-dependent quantity.

The use of fluctuation analysis for reactivity measurement has been reported;[10,11] however, these measurements were made under more or less ideal conditions (clean reactors with high detector efficiencies), and, as yet, these techniques have not been applied to shutdown-reactivity measurements in power reactors.

As stated, the quantity of interest in a noise measurement is the decay constant α related to the reactivity by

$$\alpha = \alpha_c(1 - \$)$$

where $\alpha_c = \beta/l$, the ratio of effective delayed-neutron fraction to prompt-neutron lifetime (decay constant at delayed critical), and \$ = reactivity in dollars.

The reactivity can be determined if the decay constant is obtained by measurement of either the two-detector cross-power spectral density (CPSD)[23,24] or the cross-correlation function.[25,26] Two-detector measurements are preferable because they give a zero expectation value to the white-noise components of the spectral density. The white-noise components are due to statistics of detection, as well as to backgrounds from random fission events and, for plutonium-fueled LMFBRs, backgrounds from spontaneous fission of ^{240}Pu in blankets and core. These measurements involve a finite data-processing time and, as such, do not provide an instantaneous indication of reactivity.

A reactivity meter based on the measurement of the CPSD at high and low frequencies has been reported recently.[27] The CPSDs are measured at two frequencies simultaneously, ω_L and ω_H, where $\lambda_i \ll \omega_L \ll \alpha$ and $\omega_H \gg \alpha$. The break frequency of the cross-power spectrum is designated by α, and the decay constants of the delayed-neutron precursors are represented by λ_i. The reactivity is then given by the following approximate formula:

$$\rho(\$) = 1 - \frac{\omega_H}{\alpha_c}\left[\frac{(\text{CPSD})_H}{(\text{CPSD})_L}\right]^{1/2}$$

The detector signals were filtered at high and low frequencies, and CPSDs were derived on line by appropriate analog equipment. The CPSD signals were then used as input to an analog circuit which solved the above equation. The measuring sequence was digitally controlled to allow a finite averaging time for determination of the CPSDs. The instrument does not give an instantaneous indication of reactivity, since its "updating" time depends on the averaging time required to determine the CPSDs to the desired accuracy.

The accuracies attainable are highly dependent on the detector efficiencies and the integrating time used to define the CPSDs. Kryter, Fry, and Roux[13] have estimated that, for an integration time of 15 min, a detector efficiency of 3×10^{-6} events/fission would give satisfactory results at shutdown reactivities to 5 dollars in a fast-spectrum plutonium-fueled reactor. Implementation of this technique to provide a continuous and quick indication of reactivity requires *RC* integrating time constants much shorter than 15 min, resulting in greatly reduced precision. Consequently this technique could not be applied to projected fast reactors unless marked improvements in detector efficiency could be obtained.

A reactivity meter based on the measurements of the low-frequency coherence function (correlation coefficient), reported by Seifritz,[28] does provide an instantaneous output proportional to the reactivity. The outputs of each of two detection channels were bandwidth limited to include frequency components only in the constant-amplitude region of the reactor transfer function. The bandwidth is limited so that the accepted signal from each detector has frequency components in the range $\lambda_i \ll \omega \ll \omega_c$, where ω_c is the break frequency of the reactor transfer function.

The coherence between these two signals is then strongly dependent on the system reactivity. The coherence function is calculated by circuits that compare the signs of the instantaneous detector signals.

Outputs are generated whenever the signs of the two signals are the same. This output is proportional (in practical cases where the correlation between the two detectors is small) to the coherence function.

The reactivity is related to the measured coherence functions by

$$\$ = 1 - \left(\frac{1-p}{p}\,\frac{p_c}{1-p_c}\right)^{1/2} \quad \text{in dollars}$$

The plateau values of the coherence function in the low-frequency range at the critical and subcritical conditions are p_c and p, respectively.

The output of the sign comparator is passed through a smoothing circuit to an analog voltmeter calibrated to directly read reactivity. Of all the noise techniques, this device comes closest to being a true reactivity meter. Its integration time is governed by the time constant of the smoothing network; however, accuracy is compromised for time response and vice versa. The accuracy of this technique is also compromised by low detection efficiency and is subject to systematic error whenever the detection efficiency changes,[29] i.e., in gross loading changes during refueling.

The coherence may also be measured by the same equipment used for measuring the CPSD.[30] For this purpose the outputs of the two detection channels enter the inputs of an analog multiplier. The multiplier output is integrated by a voltage-to-frequency converter having two pulse outputs, one active on positive excursions of the multiplier output and the other active on negative excursions. These outputs are counted in two separate scalers, and the difference between the scaler readings indicates the CPSD.

The ratio R between the scaler readings indicates the coherence p. These are related by the formula

$$R = \frac{\frac{2}{\pi}\left(p \sin^{-1} p + \sqrt{1-p^2}\right) + p}{\frac{2}{\pi}\left(p \sin^{-1} p + \sqrt{1-p^2}\right) - p}$$

which may be solved for the value of p corresponding to a given R.

For use at or near critical, the efficiencies attainable from practical detectors preclude use of the above noise techniques to provide an instantaneous indication of reactivity because of the extremely short measuring times allowed. However, it has been shown that adequate precision can be obtained in reactivity measurements to −3 dollars if a CPSD technique is used for measuring for 30 min.[31] Comparable precision should be attainable with these measuring times using the same detectors with the above noise techniques. However, the highly desirable feature of instantaneous reactivity indication is lost.

SUBCRITICAL MEASUREMENTS

The following subcritical reactivity-measurement methods have been proposed for use in operational fast breeder reactors, in particular the FTR. Although not satisfying the rapid-response requirements of a reactivity meter, these are alternative methods of reactivity determination applicable over the full-shutdown range of the FTR, i.e., 0 to −30 dollars.

The techniques in this category are:

1. Source-multiplication method.
2. Asymmetric-source technique.
3. Constant-rod technique.
4. Neutron-noise analysis and the low-frequency coherence technique.
5. Inverse kinetics.

Methods 4 and 5 have been discussed previously in this article. The noise analysis and inverse kinetics are somewhat limited in their range of application and depend on the attainable detection efficiency. Allowing adequate measuring times, these techniques should be applicable in the shutdown range near critical. At further subcriticalities the other techniques are more applicable, and, in principle, all are capable of measuring to −30 dollars. The low-frequency coherence-function technique discussed previously is sensitive to changes in detector efficiency and to the absolute magnitude of the efficiency. In previous applications of the technique, detector efficiencies were much greater than those attainable with practical detectors, and, as such, the method has not been fully proven in a realistic operational LMFBR application.

These methods of subcriticality measurement were recently tested in the FTR critical experiment in the ZPR-9 at ANL.[29,31-34] Of all the techniques, only the source-multiplication and asymmetric-source techniques used fission chambers with their lower detection efficiencies. Since only event counting is required for the constant-rod technique, it is concluded that this experiment can be done with similar detectors.

In the source-multiplication technique, the detector count rate is related to the reactivity by

$$CR = \frac{W S_0}{\nu \mathrm{B}\, \rho(\$)}$$

where W is the detection efficiency and S_0 the neutron-source strength. Application of the method requires a calibration at some known subcritical point. In the FTR mockup experiments, the method was calibrated by a CPSD measurement at 1 dollar subcritical. This reference point could also be established by removal of one or more well-calibrated control rods or by inverse-kinetics techniques following a moderate reactivity change. Changes in detector efficiency and source strength were accounted for by transport-theory calculations. Studies are in progress to ascertain the degree of sophistication required in these calculations.[35]

In the FTR experiments, subcriticalities measured by the source-multiplication technique agreed with those measured by other techniques and with calculated values in the far-subcritical range when corrections for W were made by calculation. Measurements[31] were made by this method over the full-shutdown range to 26.1 ± 2.1 dollars.

Another technique for subcritical measurement is the asymmetric-source technique. Placement of a large ^{252}Cf neutron source in an asymmetric location produces reactivity-dependent flux tilts. The subcritical level is inferred from detailed flux-tilt calculations and the ratio of counts obtained from symmetrically placed detectors. The experimental technique is rather simple and is quite sensitive to large reactivity changes. The primary disadvantage is the need for detailed calculations of changing reactor configurations during loading changes.

There are several methods to relate the count-rate ratios to reactivity. The first is to relate R, the detector count-rate ratios, to a simple parameter γ by

$$R = \frac{S\Delta}{(1+\gamma)\Delta - \gamma}$$

where $\Delta = 1 - \rho\beta$ (ρ is in dollars) and S is the ratio of detector sensitivities. The parameter γ can be found by computation or by calibration at a known reactivity, the simpler approach. Measurements made on the FTR critical by this method gave good results in the 0 to −2 dollar range, but overpredicted R for greater shutdown margins.[34]

An alternate method involves calculation of the relation between the λ_0 eigenvalue and R and given a measured R, a λ_0 could be inferred directly.

In the constant-rod-drop technique,[33] one or more carefully calibrated rods are inserted, i.e., a known reactivity is inserted, and the detector count rates are observed before and after the insertion. The initial reactivity is related to the count rates and known reactivity by

$$\rho_1 = \frac{\dfrac{S_1\epsilon_1 C_2}{S_2\epsilon_2 C_1}}{1 - \dfrac{S_1\epsilon_1 C_2}{S_2\epsilon 2 C_1}}\rho$$

where C_1 and C_2 = the initial and final count rates, respectively
S_1 and S_2 = the sources, respectively
ϵ_1 and ϵ_2 = the efficiencies
ρ = the known reactivity

If ρ is relatively small and represents the effects of a poison, the initial and final sources and efficiencies can be taken to be constant and the above expression reduces to a very simple form. The rods inserted during these measurements were two ^{10}B safety rods worth 82 cents and a single ^{10}B rod worth 50 cents and were calibrated by the inverse-kinetics technique at critical.

A limited test of this method was made in the FTR critical facility.[33] Measurements were made over the range of subcritical levels to ≈12 dollars with good agreement with other methods. The measurements were made using existing reactor detection channels, several of which were fission chambers, to ±5% over the shutdown range to 12 dollars.

The inconsequential changes in detector efficiency during the premeasured rod insertion and the simplicity of the method recommend its further study as a subcritical measurement technique. The same feature with slightly more complexity can also be obtained in the inverse-kinetics technique.

The pulsed-neutron technique,[36,37] in principle, is also capable of measurements down to 30 dollars. However, use of the pulsed-neutron technique requires observation of a fundamental-mode decay. In the large, heterogeneous LMFBR-type core, this may not be achievable at 30 dollars subcritical. It is not known how far subcritical one can go and still observe a fundamental-mode decay in such a core. Furthermore, ancillary problems make the technique unfeasible. To achieve this performance would require placement of the generator very close to the core, e.g., in a reflector-element position. A self-contained tube-type generator is not sufficiently reliable or resistant to the environment for long-term operation, as in continuous shutdown monitoring. Keeping the sensitive parts of the generator away from the adverse environment of the core requires the use of a drift tube. However, a drift tube of 7 ft is the longest that can be used without auxiliary focusing or accelerating elements

along the way, and that is not sufficient to remove the generator to an acceptable environment.

Recent experience[38] in pulsing a plutonium-fueled critical facility has shown that the inherent ^{240}Pu spontaneous fission source can limit the range over which the pulsed technique can be used. In these experiments it was found that the background due to this source was comparable to the signal, i.e., the multiplied accelerator produced neutrons, at a subcritical level of about −5 to 6 dollars. This limit can be extended if the pulsing is done with a stronger source. In any case the backgrounds due to the ^{240}Pu must be considered in any application of pulsed techniques in plutonium-fueled systems, particularly those which depend on the identifying of prompt- and delayed-neutron contributions to the total count rate.

DISCUSSION

In all the above-mentioned reactivity-measuring techniques, the validity of the point kinetics model has been assumed. In some cases, departures from the model are to be accounted for by space- and energy-dependent calculations. Presumably, all the techniques could be so corrected with the sacrifice of additional complexity. At or within several dollars of delayed critical and at power, detector-efficiency changes and gross flux-shape changes are not expected to be important. Consequently most of the techniques mentioned above are applicable if sufficient measuring time is allowed, with the noise techniques being least attractive because of the low detection efficiencies of the fission counters.

At far-subcritical levels the source-multiplication technique appears to be the most promising. This technique could be calibrated by one of the several methods mentioned above.

The constant-rod technique assumes that the known rod worth or reactivity insertion remains constant and the detector efficiency does not change. The first assumption is the most questionable, and further study of this problem is recommended.

The two techniques of reactivity-meter implementation, i.e., those based on noise analysis and inverse kinetics, have inherent weaknesses that can result in errors large enough to preclude their use in on-line situations. In the inverse-kinetics technique, calibration of the source term presents the most difficulty in LMFBR applications. In applications where the source term can be neglected, such as monitoring at power, this technique is clearly preferable and its applicability and utility have been sufficiently demonstrated.

The suitability of the noise techniques depends on the available detector efficiency and the time allowed for the reactivity determination. The detection efficiency obtainable in practical detectors precludes using these methods to provide rapid indication, although reactivity information could be obtained with considerably longer measuring times.

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Reactor-Noise Monitoring for Malfunctions

By Joseph A. Thie*

Abstract: *Utilization of randomly fluctuating signals from reactor instrumentation for extracting useful information is reviewed. Emphasis is placed on ability to detect abnormalities. Signal sources treated are neutron flux, temperature, vibration, sound, strain, pressure flow, and gamma flux. Methods of analysis are discussed within the current state of the art.*

With the rising interest in the mechanical integrity of structures within the reactor vessel, more-demanding quality-assurance programs are now being used in the manufacture of well-designed reactor components. Once operation has commenced, however, it becomes difficult to continue assurance of the somewhat related goals—integrity of the hydraulic and mechanical systems. The specific kinds of operational monitoring that can be most effective are reviewed here.

REASONS FOR MONITORING

Interest in this type of monitoring has been encouraged since 1969 by the Advisory Committee on Reactor Safeguards (ACRS). In the ACRS review of license applications of large water reactors, a condition for approval was that the applicant would have a satisfactory program to monitor for vibrations and loose parts. This concern can be justified by the significant number of incidents involving the loss of core mechanical integrity or fuel-element cooling.

Listed in Table 1 is an assortment of these incidents with varying amounts of damage; it is significant that forewarning of all incidents was given by a randomly fluctuating (i.e., noisy) variable. It might be presumed that an adequate surveillance program (i.e., appropriate instrumentation combined with operator attention) would have prevented the worst cases where fuel melting occurred.

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Generalizing somewhat from the examples of Table 1, we can list these potential mechanical/hydraulic malfunctions within the reactor vessel:

1. Fatigue or cracks in the metal of the vessel, internal structure, or piping.
2. Bolts or other means of fastening which have come loose.
3. Wearing away of metal.
4. Control-rod-movement abnormalities.
5. Flow blockage caused by accumulations, foreign materials in the system, or structure that has broken loose.
6. Excessive vibration.
7. Instabilities or other departures from normal cooling.

Although not intended to be complete, this list is realistic in that all categories are based on actual experiences.

The philosophy behind operational monitoring for such malfunctions reviewed here is a logical extension of that used in reactor design, component fabrication, and final assembly. Figure 1 shows various efforts aimed at minimizing the probability of difficulties from the malfunctions listed. To omit the final step shown—monitoring during operation—would not be consistent with the extent of effort previously expended. So monitoring during operation should be used to reduce the probability of difficulties to well below that implied by the number of examples in Table 1.

The open-vessel work over a reactor core illustrates the exceptional concern for core integrity; the workers are most sensitive to the possibility of foreign objects remaining in the vessel. Thus concern about loose objects after the vessel is closed should not be surprising. Monitoring activities specifically oriented to assurance in this area are considered very important.

Table 1 Reactor Incidents* Due to Mechanical or Hydraulic Difficulties Within the Primary System

Malfunction class	Reactors having the malfunction: With fuel melting	Reactors having the malfunction: Without fuel melting	Specific cause	Parameter whose noise changed
Object foreign to vessel causing flow blockage	ETR		Piece of plastic[9]	Neutron flux
	SRE core 1		Pump-coolant decomposition products[20]	Exit thermocouple
	MTR		Debris from gasket[10]	Automatic-regulating-rod position
	ORR		Neoprene gasket[11]	Neutron flux
Object intrinsic to primary system loose and causing flow blockage		Pathfinder	Steam-separator pieces in a recirculation line[17]	Recirculation flow
		Windscale AGR	Duct inner-insulation collapse[48]	Pressure
	Fermi		Loose plate at core inlet[12]	Neutron flux
Vibrating and loose, but still intact, vessel parts		Big Rock, Trino, and Obrigheim	Loose thermal shield[5]	Neutron flux
		ETR	Vibrating tube of a test loop[49]	Neutron flux
		GETR	Loose control rods from wear[8]	Neutron flux
		SRE core 2	Loose fuel[13]	Neutron flux
		HFIR	Control-rod-bearing failure[3]	Neutron flux

*All these incidents required shutdown and repair, and forewarning of the incidents was given by noise monitoring.

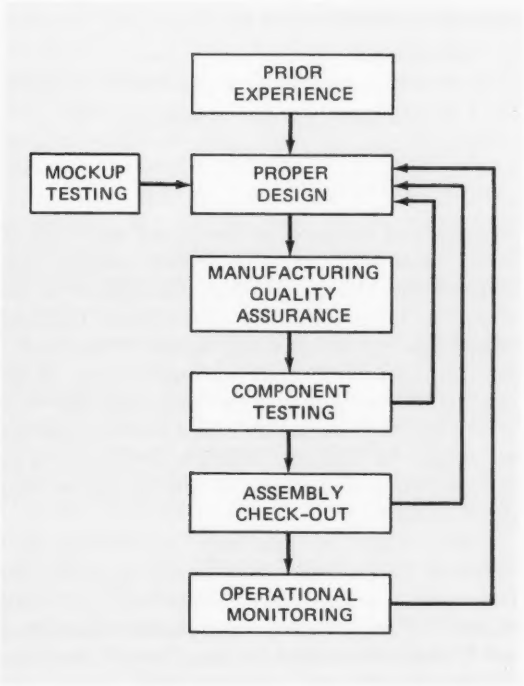

Fig. 1 Sequential steps taken to ensure core mechanical/hydraulic integrity. Sources of experimental feedback to the designer are indicated.

In regard to incentives for monitoring, the nuclear power and airline industries are somewhat analogous: both have a complex and expensive product that must be capable of reliable operation with a minimum of downtime; also, both are safety conscious and heavily regulated. In the older aircraft industry, precedents can be found in the use of operational monitoring to enhance mechanical reliability. Among techniques operators have found beneficial is "listening" for noises in an automated sophisticated fashion. A strong incentive comes from being able to extend the periods between maintenance shutdowns. This is because the Federal Aviation Agency is willing to permit longer intervals between engine overhauls if there is adequate operational vibration monitoring.

INFORMATION TIME SCALES

The impressive diversity of instrumentation used by the operators of present-generation power plants is primarily oriented at displaying quasi-steady-state values of many parameters rather than being concerned with the normally occurring fluctuations in process parameters. This review considers how lower thresholds of malfunction detection can be achieved when high-frequency information in parameter fluctuations is used in addition to the volume of quasi-steady-state values of the variables observed.

In the broad picture of information transmittal from the interior of a reactor vessel to the operator, the data or intelligence can be classified according to its characteristic times. Table 2 arbitrarily selects decades, and its examples indicate somewhat how characteristic times are defined. Perhaps a qualitative definition of the characteristic time of a data type would be the minimum length of measuring time required to perceive its basic information content. For example, xenon transients observed over a 10-hr interval permit some measurement of the iodine-to-xenon decay time constant inherent in the process.

Table 2 Times Associated with Reactor Dynamic Processes

Approximate range of the characteristic times, sec	Examples of the dynamic processes
10^3-10^5	Xenon; vessel cooling
10^2-10^3	Internal structure cooling
$10-10^2$	Automatic controllers of process systems
0.1–10	Coolant transport; heat transport
0.01–0.1	Vibration of internals
$10^{-4}-10^{-2}$	Neutron lifetime in thermal reactors; metallic sounds
$<10^{-4}$	Neutron lifetime in fast reactors; stress-wave emissions

A more fundamental definition for this characteristic time is the dominant parameter with dimensions of time that appears in the mathematical description of a dynamic process. The 6.7-hr half-life of iodine-to-xenon decay is an example. Often, more than one such time constant may be important. If time constants are widely spaced, their separate measurement and study are conceptually possible; if closely spaced, a single average may tend to be manifested.

With this broader outlook on data associated with reactor operation, additional dimensions in monitoring for malfunctions appear. To the N signals (N-dimensional monitoring) whose steady-state values are routinely monitored can be added $\Sigma_{i=1}^{N} m_i = \bar{m}N$, where m_i is the number of distinct higher frequency phenomena in the ith signal ($\bar{m}N$-dimensional monitoring). Each of these phenomena might occur in a frequency band, $2\pi(f_{hi} - f_{lo})$, centered around a value equal to the reciprocal of a time constant. These are frequency "spectral windows" through which each signal can be observed: the lowest is steady-state observing (e.g., average fission-chamber current), and higher windows are determined by system time constants (e.g., local boiling noise at $2\pi f$'s near the reciprocal of bubble residence times).

Determining the relative utility of monitoring these various windows in various signals is not simple. Extensive operational experience in combination with interpretive analysis is required. So far, experience is leading interpretation somewhat. The following sections summarize this experience to date, a variety of detectors having demonstrated potential utility in many spectral windows. It will be apparent that restricting monitoring activities merely to steady-state signal values is somewhat shortsighted.

NEUTRON FLUX

Changes in neutron flux can be used to monitor local or remote conditions, depending on the physics and geometry involved. Specific factors that determine the entire detection process are

1. Reactivity.
2. Fuel and transmission region for fast neutrons.
3. Moderation region near the detector.
4. Absorption near the detector.
5. The detector itself.

Mediums and structures are associated with each of these. Observing the neutron-flux changes can therefore monitor the significant contributors in the chain 1 to 5. But whether, in fact, a desired threshold of detection of a particular malfunction or abnormality can be achieved depends on such factors as its quantitative contribution to the fluctuations relative to that of other factors, and the extent to which measures are taken to enhance detection sensitivity (e.g., choosing the optimum spectral window and the best detector location).

The causes of reactivity fluctuations that ultimately lead to detector fluctuations are diverse and have been quantitatively understood to some extent.[1,2] Fuel, moderator, and absorber geometrical and density effects lead to many specific reactivity coefficients, dk/dv, and thus contributions to reactivity

$$\delta k = \frac{dk}{dv} v(t) \qquad (1)$$

where $v(t)$ is a specific independent variable that may contribute a fluctuating input within the core. It can represent, for example, the vibration of control rods, or perhaps local temperature or steam-void fluctuations.

A few cases are singled out here as specific examples of these effects in specific reactors. In the High-Flux Isotope Reactor (HFIR) it was possible to monitor the wear of bearings in the control-rod drives.[3] Hydraulic instabilities in early boiling-water reactors (BWRs) have been detected[4] and even predicted[5,6] during escalation of power. In the pressurized-water reactor (PWR), the effect of coolant chemistry changes on the fuel has even been studied by neutron noise.[7]

An important chain in the use of reactivity noise for monitoring coolant-flow distribution is: obstruction or modulation → temperature or steam voids → reactivity. Examples exist in which feedwater distribution into recirculating water has spontaneously changed,[2] a loose and vibrating thermal shield has modulated flow,[8] and loose parts or foreign debris have blocked fuel channels.[9-12] In each instance the neutron fluctuations gave an early warning in the course of the incident.

Next in the chain between reactivity and the detector are the fuel and other regions through which the fission neutrons travel. In general, fuel motion—such as loose fuel rods in the Sodium Reactor Experiment (SRE)[13]—would be in the category of reactivity noise effects just discussed. However, the combination of peripheral fuel, downcomer coolant, and intervening structures represent an attenuation path for fast neutrons toward out-of-vessel detectors. Changes here can be monitored to the extent they significantly modulate the neutron signal. In the La Crosse Boiling-Water Reactor (LACBWR),[14] the external detector noise is a very sensitive indicator of conditions in the two-phase mixture in steam separators surrounding the core. In Trino,[15,16] loose and damaged structures were detected by neutron noise in external detectors, and residual motion—only a few hundredths of an inch vibration—of the repaired internals is now routinely monitored for any changes.

Another means by which the neutron flux indicates a specific hydraulic or mechanical condition is by local moderation and/or absorption near the detector. The ratio of high- to low-frequency noise in BWRs increases as the local steam-void content increases,[17] due to a high-frequency modulation of the flux signal by the moderating process. In addition to monitoring local hydraulic conditions, an in-core detector can follow small motions of nearby control rods, 0.001-in. amplitude being a threshold of detection reported by Rajagopal et al.[18]

Finally, the detector itself can offer a noise contribution that can be more bothersome than useful, depending on its magnitude and cause. The random arrival of neutrons is always causing chamber current fluctuations. Also, an ion chamber located in a large flux gradient is sensitive to small mechanical vibrations: air-hammer operation on concrete in a building adjacent to the Penn State reactor has been observed in the noise spectrum of an ion chamber.[19]

Regarding the types of neutron detectors in current use—ion chambers, fission chambers, and self-powered detectors—the useful upper frequency limit is an important consideration. Response at acoustic frequencies is readily possible with ion and fission chambers. But some self-powered detectors, depending on their beta-decay half-life (of the order of a minute for some), are of rather limited value for the monitoring types discussed here if only very low frequencies can be followed. A virtually prompt cobalt type of self-powered detector (based on internal conversion) does not have this disadvantage, however.

TEMPERATURE

Local coolant temperature is an excellent indication of hydraulic conditions. For thermocouples located at the exit of fuel channels, there is a direct measure of possible flow blockage. In addition to the steady-state values of temperature at a given power, the local temperature fluctuations offer further monitoring possibilities. A striking change in the character of exit temperature noise occurred in the blockage of an SRE fuel channel.[20] Boiling studies by Fry, Kryter, and Robinson[21] also indicated that the character of cladding thermocouple noise is a very sensitive indicator of channel boiling.

Such use of core exit thermocouples in the appropriate spectral window can be either power monitoring or flow monitoring. This is because of the relation

$$\delta T \sim \delta\left(\frac{\text{power}}{\text{flow}}\right) = \frac{\text{power}}{\text{flow}}\left(\frac{\delta\ \text{power}}{\text{power}} - \frac{\delta\ \text{flow}}{\text{flow}}\right) \qquad (2)$$

where δ refers to random fluctuations about average values. Either term on the right may dominate.

Also, thermocouples elsewhere in the vessel can be used as flow-distribution indicators by virtue of the transport of a random distribution of "lumps" of coolant having slightly different temperatures as they pass by the thermocouples. In this case

$$\delta T \sim \text{mixing length} \times \text{overall gradient of the average temperature field} \quad (3)$$

In the Elk River BWR, the noise of thermocouples below the core was found to be correlated with neutron noise behavior.[2] In another application, Randall and Pekrul[22] have been able to measure local flows and even to determine their spiraling pattern by cross-correlating fluctuations of separated pairs of thermocouples in the coolant stream as these sequentially detect the same "lumps."

Where a measurable cross-correlation of temperature and flux noise signals exists, such as in the Dounreay Materials Testing Reactor[23] and Fermi,[24] this becomes a much better monitor than either signal by itself. Changes in the correlation from its norm can be due to factors that disturb the neutron flux, the coolant temperature, and any intermediates involved in their correlation.

VIBRATION

The most direct method of measuring vibration of reactor internals is to install motion transducers on suspect components, preferably near points of maximum expected vibration. These transducers can be linear differential transformers, velocity sensors, or accelerometers—measuring displacement, velocity, and acceleration, respectively. Unlike most other reactor variables, these are vectors: to specify completely the motion of a point requires three mutually perpendicular measurements. In practice, however, a dominant vibration mode is predicted, which may be sufficient to measure only one or two directions. It is also sufficient to measure displacement $x(t)$, velocity $v(t)$, or acceleration $a(t)$ because of their mutual relations:

$$x(t) = \int v(t)\, dt = \iint a(t)\, d^2t \quad (4)$$

All three types of sensors have been used in the adverse temperature and radiation environments of water reactors.[25] Both the differential transformer and the velocity sensor use magnetic coupling, whereas a piezoelectric crystal is used in the accelerometer. All have adequate frequency response for measuring typical core-component vibrations (up to 100 Hz is usual).

Proper location of these transducers on reactor internals is the key to adequate monitoring. Selections are made on the basis of

1. Mathematical models of vibratory motion of the core internals.

2. Experimental studies of vibration on full- or reduced-scale models.

3. Prior experience in similar reactors.

In mathematical modeling, it is feasible to calculate the dominant frequencies and corresponding directions of vibration for all major structures within the vessel. In the usual undamped mode calculation, based on lumped masses and equivalent coupling springs, it is not possible to calculate absolute amplitudes. This is because the amplitude calculation depends on having a detailed knowledge of the excitation forces (from the flowing coolant) and damping forces. However, the ability to calculate relative amplitudes at various points for all the structures is sufficient, when combined with an experimental measurement of the amplitudes, to describe peak stresses and hence limiting criteria of metallurgical fatigue.

Table 3 indicates the kinds of vibration-monitoring programs adopted during the startup of BWRs. Of these, the KRB program[25] was the most extensive. It was demonstrated that a detailed mathematical model can accurately predict the observed frequencies of the individual resonant structures. But the data revealed some as-built aspects of the assumed structural detail as capable of affecting vibration results.

Vibration testing of vessel internals is performed at various flow and temperature conditions during reactor commissioning. Also, to verify that design margins (with respect to fatigue) are always met, tests are conducted during unbalanced and transient flows, where excitation of structure resonances is more likely. However, after startup testing is complete, an incentive to continue monitoring can remain—stemming from a desire to detect structural changes (loosening of bolts, for example), which may subsequently cause operational problems.

As noted above, in contrast to local vibration monitoring, gross monitoring is possible through neutron-flux fluctuations. Although it requires special experimental conditions, another technique, developed by Randall and Pekrul,[22] should be mentioned: coolant conductivity fluctuations between probes on fuel elements detected 0.0001-in. amplitudes.

Table 3 Vessel-Structure Vibration-Monitoring Programs

Reactor	Number of transducers: Acceler-ometers	Velocity sensors	Differential transformers	Strain gauges	Locations of transducers
KRB	6				Diffuser, baffle
	4				Core support plate
	4		8		Separator, dryer
				4	Control-rod-guide tubes
				2	In-core flux tubes
				8	Fuel channels
Tarapur	3				Diffuser, baffle
			4		Separator, dryer
			3		Thermal shield
Nine Mile	2				In-core flux tube
Point	4				Outer surface of vessel
Dresden 2				4	Control-rod-guide tubes
				4	In-core flux tubes
				8	Fuel channels
	3				Core support plate
			4		Shroud
			4		Separators
			17		Jet pumps
Monticello		4			Separators
			4		Shroud
			4		Jet pumps
				4	Control-rod-guide tubes
	4				In-core flux tubes

SOUND

A distinction is made here between vibration and sound in their applications to reactor monitoring. Vibration is characterized by specific components moving in specific directions, with low frequencies (such as below 100 Hz) usually but not necessarily prevailing. Also, vibration is more likely to be measured at the source, whereas a transmission medium is implied with sound.

As might be expected, a prime contributor to the acoustic spectrum, whether in water-cooled or liquid-metal-cooled reactors, is coolant noise. This includes noise from the pump itself as well as from the turbulence of the flowing coolant. Advantages stem from such noises when a direct monitoring of pumps and coolant is desired. However, this background is a disadvantage in monitoring other acoustic phenomena, such as sounds from loose parts involving intermittent metallic contact. The threshold of detection for the latter is raised in proportion to the spectral content of this background noise within the frequency window being used. But, if correlation techniques are used (as between two pickups), this threshold of loose-parts detection can be significantly lowered because the unwanted random contributions do not appear in the correlation.

Even if background complications are overcome, the conceptual simplicity of sound monitoring for loose parts and other mechanical abnormalities can be misleading. Acoustic transmission paths through a mixture of media are complex. Moreover, the reactor vessel and associated piping themselves are resonant structures. For these reasons the present state of the art requires experimentation to seek out optimum listening locations for the transducers. Furthermore, unless provisions are made in the design phase for the core structure and the vessel, it may be left to chance whether suitable sound-monitoring locations can be found later.

Accelerometers are the transducers typically used in sound monitoring of reactor structures. Exterior points of attachment are selected to permit changing transducers when necessary. These may be the vessel wall, a pipe, control-rod structure, or an instrument thimble. But, instead of these structures as wave

guides, a special metal rod was used[26] in the Dounreay Fast Reactor for coupling the primary coolant to a wide-frequency-band accelerometer located away from the reactor environment.

Some examples might be cited of the variety of experience in reactor sound monitoring. Following the coolant-blockage incident in the Fermi reactor, acoustic monitoring tests under various flow-transient conditions were performed to detect loose metallic pieces in the bottom of the vessel.[10] But, because of the lack of earlier comparison data in the absence of loose pieces, these tests were not conclusive.

In the Experimental Breeder Reactor II (EBR-II) plant, there is a current application of routine acoustic monitoring by Anderson and Just[27] and Price and Karvinen.[28] Accelerometers at various locations outside the primary- and secondary-system coolant boundaries are used to record flow noise at various operating conditions. Comparison of sound spectra—normal (before a small pipe within the intermediate heat exchanger became loose) with abnormal (after this pipe was loose)—was found to be a valuable tool.

Gas bubbles have been detected in the SRE coolant with accelerometers.[29] Acoustic detection of boiling has been the subject of a general review by Saxe[30] and a liquid-metal-oriented review by Anderson, Mulcahey, and Hsu.[31] Emission frequencies of interest typically range from 1 to 30 kHz. The phenomena monitored here are also related to that of pump cavitation: collapse of small bubbles in a subcooled liquid, with sonic frequencies varying inversely with bubble size. In contrast to passive monitoring, active methods have been proposed by DePrisco et al.[32] Ultrasonic excitation of a potentially boiling fluid is detected, with reflected energy sensed being responsive to local conditions of incipient boiling.

An encouraging step toward achieving a practical means of monitoring for loose pieces in an operating PWR was experienced at Trino.[16] With 13 accelerometers located on the primary pipes and vessels, it was found that a loose piece scraping along a primary-system wall can be detected in the 200- to 2000-Hz range of frequencies monitored. Hence operators listen to these accelerometers once a day and record their signals once a week.

Ultrasonic monitoring also has received some attention, such as when there is an aim of locating coolant leaks by the escaping-fluid sounds transmitted through the piping and vessel-wall paths. In the Elk River reactor,[33] it was possible to triangulate toward the location of a pipe crack by sound-transmission time correlations among pickups.

In other ultrasonic applications concerning primary-system monitoring, stress-wave emissions around 10^5 to 10^6 Hz increase and forewarn cracking failures. Practical applications of this have been demonstrated on the SM-1A vessel,[34] and on-line monitoring is now believed to be practical.[35]

STRAIN

Vibrations of core internals have been successfully monitored by the use of strain gauges.[25] The simplicity of resistance elements allows transducers to perform satisfactorily in adverse radiation and temperature environments. As with other direct vibration measurements, the direction as well as the magnitude (usually expressed as microinches of strain per inch of length) is significant. Also, the points of maximum strain are, of course, different from those of maximum amplitude. Thus, placement considerations discussed for vibration motion sensors also apply here.

An outstanding application of strain gauges was in the Halden BWR,[36] where it was possible to measure the cyclic elongation of fuel-element cladding during vibrations. In SPERT-III,[37] it was possible to study thermowell fatigue due to flow-induced vibrations, thereby preventing reoccurrence of a primary pipe's thermowell breaking loose and entering the vessel.

PRESSURE AND FLOW

Pressure, either absolute (i.e., psig relative to atmospheric pressure) or differential (e.g., local difference between two points, such as across an orifice), can be a means of monitoring hydraulic conditions. A comparison of these pressures shows that the differential pressure is a more direct and localized measurement influenced by flow; on the other hand, the overall pressure is a general thermodynamic indicator of the entire coolant. In both kinds of pressures, there can be potential in information from higher frequencies not now routinely monitored. But local pressure differences have large potential because of the capability of monitoring specific areas.

A systematic spatial mapping of fluctuations in differential pressures can contribute substantially to a fundamental understanding of hydraulically induced vibrations. Years of theoretical and experimental effort give reactor designers an accurate quantitative knowledge of steady-flow values throughout the vessel. A review by Wambsganss[38] reported some knowledge of the higher frequency content of flow fluctuations, but it is quite limited, as is its use in predicting amplitudes

of hydraulically excited vibrations. Measurement of the frequency content of the hydraulic energy of the fluid at key points, if combined with associated vibration measurements of structures, would be a means of improving the fundamental understanding of reactor structure vibrations.

In the Pathfinder reactor, a broken piece from an internal steam separator first manifested itself by minor changes in the steady value of recirculation flow and hence was not of major concern to the operators. But, at the same time, major changes in the frequency spectrum of the recirculation flow fluctuations were discovered.[17] Thus the character of differential pressure fluctuations can be sensitive to flow obstructions even if these should cause little or no change in the average value of the flow.

Where reactor pressure changes are related to reactivity changes, as in BWRs or in nonboiling reactors near the threshold of boiling, a meaningful cross-correlation can be found. Experiments at the HFIR[21] investigated the use of this cross-correlation to detect the onset of boiling. Of practical value to PWRs is the experience of Trino:[16] the neutron and pressure noises are correlated at frequencies near a dominant 4.5 Hz, which is of possible mechanical origin in the core. In addition, the absolute pressure noise alone provided a suitable signature for a general monitoring of the primary system and, in particular, was sensitive to metal-on-metal noises (such as the operation of control-rod magnetic jacks).

GAMMAS

Kenney[39] has experimentally demonstrated that fluctuations of prompt-gamma emissions from the core have essentially the same characteristics as the neutron noise. Osborn's theoretical work[40] indicates that the range of effective monitoring in very large reactors is larger for gammas than for neutrons. This means that detectors outside the vessel can "see" further into large cores by monitoring prompt gammas than by the usual neutron monitoring.

In the LACBWR reactor, comparisons were made of the gamma and neutron noise using out-of-vessel detectors. Transmission from the core through a downcomer having fluctuating steam present was not quite the same for gammas and neutrons because of different attenuations, but either signal could provide a monitoring of downcomer hydraulics from outside the vessel. Even the effect of the reactor water level on steam-bubble carry-under could be studied, as seen in Fig. 2.

A noise analysis of ^{16}N radiation in the risers carrying a two-phase mixture from the Dresden 1 vessel was performed by Beckjord.[41] Small oscillatory flow pulsations were observed and were found to be a driving force for the neutron noise in in-core chambers.

It is likely that using gammas for purposes of monitoring structural integrity within the primary system will be less prevalent than other methods discussed above. This is not due to any fundamental reason but to the very limited use of gammas so far, and because other direct approaches are available.

SIGNATURE ANALYSES

Concern about mechanical/hydraulic integrity exists during reactor startup because of possible design or construction inadequacies. But a continued concern throughout reactor life can exist because of possible unexpected malfunctions. An answer to the former concern is the special testing effort during reactor commissioning. But subsequently during the life of the plant, rather than continuing to repeat many tests, it is more practical to use some form of signature analysis. As the name implies, certain simple tests which give a sensitive and overall description of normal conditions are selected with the idea that they are as repeatable in results as is a person's signature.

Mechanical signature analysis (MSA) is a well-established tradition in industry for certain types of equipment. Early detection of bearing failures is a typical application. The self-excited signatures of operating equipment are employed more commonly than the more informative (but more complex to perform) response on a shaker.

An elementary and primitive form of signature analysis in reactors is a very valuable type of visual and aural monitoring by the operators. Unofficially, operators note the width of ink traces on the neutron-flux, primary flow, etc., recorders from day to day for similiar reactor conditions. For slow chart speeds, this indicates whether the tails of amplitude probability distributions have changed. In making inspections of the plant, sounds of pumps, for example, are characterized as normal if there are no significant changes from day to day. In this case an acoustical spectrum is the signature used.

More sophisticated signature methods with a higher sensitivity for detection of malfunctions are available, of course. Most common, perhaps, is the frequency spectrum of the fluctuations of a signal. A priori understanding of the shape of this function may be (and usually is) lacking; nevertheless, normal behavior

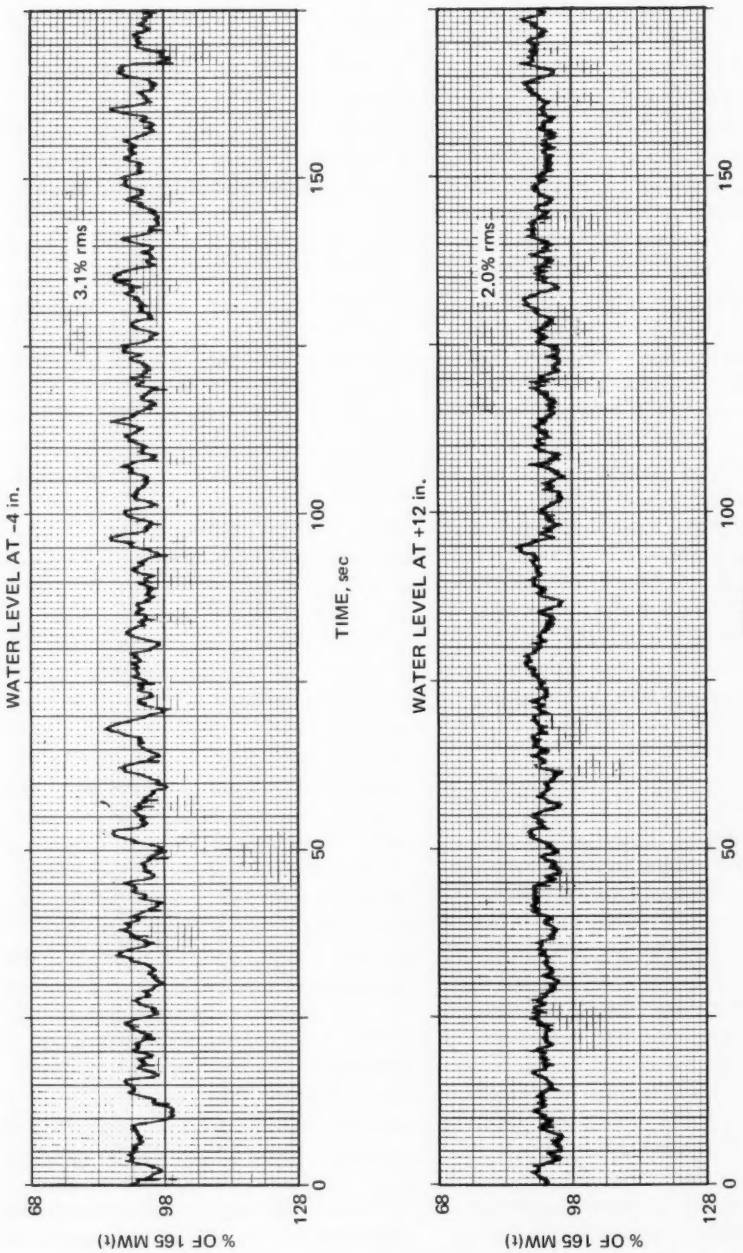

Fig. 2 An example from the LACBWR noise studies of how observing noise under different reactor conditions gives information as to its cause: the higher water level here gives better separation of steam from recirculating water, and the reduced carry-under of bubbles means less attenuation noise for neutrons transmitted through the downcomer to external detectors.

is associated with a specific shape for a given operating condition. If changes in noise occur at certain frequencies, this is an indication of off-normal behavior—with the frequencies at which a change was noticed giving evidence of the type of abnormal behavior.

Many other functions[42,43] are also suitable for signatures. The probability distribution of fluctuation amplitudes is one. Another is the autocorrelation function, which displays the equivalent information in the time domain that the frequency spectrum displays in the frequency domain. This function shows the extent of the correlation of a signal with itself at various time lags between signal values. Where a correlation between two signals is established, it can be used as a signature in the time domain as a cross-correlation function and in the frequency domain as a cross-spectrum. It is especially advantageous to use cross-correlation when small signals of interest are to be monitored in the presence of a background of much larger signals. A special correlation investigated by Cohn[44] is advantageous for on-line monitoring using the simplicity of polarity correlation (i.e., comparing only the sign of fluctuations about a zero mean value) in a high-speed digital computer.

The accuracy (and hence sensitivity for detecting changes) of a given signature-analysis technique can be enhanced in a number of ways:

1. Properly selecting transducers and their locations.
2. Accepting a broad rather than a narrow frequency resolution throughout the frequency band measured.
3. Using a long measuring time.
4. Finding, if possible, high frequencies rather than low frequencies, which are indicative of the malfunctions to be monitored.
5. Using correlation and frequency selectivity to eliminate unwanted background signals.

For continuous on-line monitoring, applying as many of these methods as possible is especially advantageous. This then allows short measuring times (for a given accuracy) and hence earlier warnings of abnormalities.

Reactor operating staffs are being encouraged by the ACRS[45] in the use of signature analysis, with such variables as the neutron flux being used. A program of taking signatures has been in effect at EBR-II for some time.[27,28] A justification for this effort by the staff is the assistance in diagnosing problems and, hopefully, of reducing downtime. Tape recordings of random fluctuations are recorded and spectrally analyzed for core neutron flux and acoustic noises from various sources (such as the primary pumps, the intermediate heat exchanger, and the steam superheater). Another example of regular signature analysis is at the Point Beach reactor.[46] Signals spectrally analyzed from 1 to 600 Hz are neutron flux, flow, pressure, temperature, and power supply.

DETERMINING NOISE CAUSES

For maximum utility from a noise-monitoring effort using any of the noise signals discussed above, it is advantageous to have at least a qualitative understanding of the cause. (As technology progresses, one would even expect to have quantitative understandings.) Evidently this is a step beyond the simpler philosophy of watching for uninterpreted signature changes.

How one might determine sources of noise can only be specified in rather general terms here, because each reactor (of a given type and generation) and each type of noise are different. Among the methods that have been successfully applied are:

1. Postulating mathematical models for a cause and comparing model predictions with measurements.
2. Depending on basic research in similar reactors or special test loops.
3. Finding correlations between different noise signals from a common source.
4. Observing the effects of various reactor conditions on the character of the noise (for example, Fig. 2).

When it is possible to bring some combination of inductive (1 and 2) and deductive (3 and 4) methods to bear on the problem of determining a specific noise source, it is reasonable to expect a satisfactory solution in most cases without expenditure of unreasonable analytical efforts. The existing literature on power-reactor noise does not suggest this, however. This may be because in the large majority of cases the program was essentially experimental only—rather than a balanced program of data collection and its analytical diagnosis.

DISCUSSION

A variety of monitoring methods have been reviewed and appraised. It is appropriate now to inquire which of these are the best. The following questions are pertinent:

1. What and where are the areas of concern for which monitoring is desired?

2. Within existing technology and experience, what are the more suitable methods for these identified areas?

3. What technological advances might be made and would be suitable for application in the near future?

Answers to the first question come from designers and design reviewers and can be expected to vary in specifics from plant to plant. Categories currently being mentioned are vibration of vessel internals or other components, loose pieces within the vessel or primary coolant loops, and flow interference by any means.

To shed some light on the second question, see Table 2 for some typical vibration-monitoring programs used in BWR startups over a period of years. Note that a diversity of transducers is still a preferred approach to vibration measurement in localized areas. Without allowances for this instrumentation in the original design, the transducers must currently be located on a best-effort basis.

Detecting loose fragments in the coolant can be done sonically or by their effect on the flow. The former is direct but difficult because of background sounds. Developmental effort in a number of reactors would be required to demonstrate practicality and should be encouraged.

The use of relatively easy but, unfortunately, less direct means of monitoring for structural vibrations as well as loose pieces and flow obstructions are well worth considering. There are long-proven methods of utilizing fluctuations in normal instrumentation (especially the neutron flux) as indicators of malfunctions. No extrapolations of technology are involved, and the signals are available throughout the life of the plant. At present, this approach cannot readily pinpoint specific locations or quantitatively measure the noise source amplitudes. On the other hand, it is quite broad in two respects: the types of abnormal behavior monitored and the extensive regions covered.

A reasonable monitoring program for a reactor at the present time would obviously center around three efforts: using direct and local vibration indicators, using indirect but broadly indicating fluctuations of normal instrumentation, and contributing to the fundamental development of these and other noise-monitoring methods from both a theoretical and an experimental standpoint. In the latter case, suggestions have been made in the literature.[1,47]

Finally, a possible objection to the type of monitoring discussed here should be mentioned. There can be concern on the part of the operating staff that unnecessary shutdowns might result from uncertain interpretation of the data. This can be reckoned with by having the proper advance criteria, making knowledgeable interpretations, and having sufficient operating experience with noise-monitoring methods to generate increased confidence.

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